APPENDIX A

TO

OPERATING LICENSE DPR-28

TECHNICAL SPECIFICATIONS

AND BASES

FOR

VERMONT YANKEE NUCLEAR POWER STATION

VERNON, VERMONT

VERMONT YANKEE NUCLEAR POWER CORPORATION

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1.0 DEFINITIONS

The succeeding frequently used terms are explicitly defined so that a uniform interpretation of the specifications may be achieved.

A. Reportable Occurrence - The equivalent of a reportable event which shall be any of the conditions specified in Section 50.73 to 10CFR Part 50.

B. Alteration of the Reactor Core - The act of moving any component affecting reactivity within the reactor vessel in the region above the core support plate, below the upper grid and within the shroud. Normal movement of control rods or neutron detectors, or the replacement of neutron detectors is not defined as a core alteration.

C. Hot Standby - Hot standby means operation with the reactor critical and the main steam line isolation valves closed.

D. Immediate - Immediate means that the required action will be initiated as soon as practicable considering the safe operation of the unit and the importance of the required action.

E. Instrument Calibration - An instrument calibration means the adjustment of an instrument signal output so that it corresponds, within acceptable range and accuracy, to a known value(s) of the parameter which the instrument monitors. Calibration shall encompass the entire instrument including actuation, alarm, or trip. Response time as specified is not part of the routine instrument calibration but will be checked once per operating cycle.

F. Instrument Check - An instrument check is qualitative determination of acceptable operability by observation of instrument behavior during operation. This determination shall include, where possible, comparison of the instrument with other independent instruments measuring the same variable.

G. Instrument Functional Test - An instrument functional test shall be:

1. Analog channels - the injection of a signal into the channel as close to the sensor as practicable to verify operability including alarm and/or trip functions.

2. Bistable channels - the injection of a signal into the sensor to verify the operability including alarm and/or trip functions.

H. Log System Functional Test - A logic system functional test means a test of all relays and contacts of a logic circuit from sensor to activated device to insure all components are operable per design intent. Where possible, action will go to completion, i.e., pumps will be started and valves opened.

I. Minimum Critical Power Ratio - The minimum critical power ratio is defined as the ratio of that power in a fuel assembly which is calculated to cause some point in that assembly to experience boiling transition as calculated by application of the appropriate NRC-approved critical power correlation to the actual assembly operating power.

J. Mode - The reactor mode is that which is established by the mode-selector-switch.

1.0 DEFINITIONS

K. Operable - A system, subsystem, train, component or device shall be operable or have operability when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electrical power sources, 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).

L. Operating - Operating means that a system or component is performing its intended functions in its required manner.

M. Operating Cycle - Interval between the end of one refueling outage and the end of the next subsequent refueling outage.

N. Primary Containment Integrity - Primary containment integrity means that the drywell and pressure suppression chamber are intact and all of the following conditions are satisfied:

1. All manual containment isolation valves on lines connecting to the reactor coolant system or containment which are not required to be open during accident conditions are closed.

2. At least one door in each airlock is closed and sealed.

3. All automatic containment isolation valves are operable or deactivated in the isolated position.

4. All blind flanges and manways are closed.

O. Protective Instrumentation Definitions

1. Instrument Channel - An instrument channel means an arrangement of a sensor and auxiliary equipment required to generate and transmit to a trip system a single trip signal related to the plant parameter monitored by that instrument channel.

2. Trip System - A trip system means an arrangement of instrument channel trip signals and auxiliary equipment required to initiate action to accomplish a protective trip function. A trip system may require one or more instrument channel trip signals related to one or more plant parameters in order to initiate trip system action. Initiation of protective action may require the tripping of a single trip system or the coincident tripping of two trip systems.

3. Protective Action - An action initiated by the protection system when a limit is reached. A protective action can be at a channel or system level.

4. Protective Function - A system protective action which results from the protective action of the channels monitoring a particular plant condition.

P. Rated Neutron Flux - Rated neutron flux is the neutron flux that corresponds to a steady state power level of 1593 thermal megawatts.

Q. Rated Thermal Power - Rated thermal power means a steady state power level of 1593 thermal megawatts.

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R. Reactor Power Operation - Reactor power operation is any operation with the mode switch in the "Startup/Hot Standby" or "Run" position with the reactor critical and above 1% rated thermal power.

1. Startup/Hot Standby Mode - In this mode the low turbine condenser volume trip is bypassed when condenser vacuum is less than 12 inches Hg and both turbine stop valves and bypass valves are closed; the low pressure and the 10 percent closure main steamline isolation valve closure trips are bypassed; the reactor protection system is energized with IRM neutron monitoring system trips and control rod withdrawal interlocks in service and APRM neutron monitoring system operable.

2. Run Mode - In this mode the reactor system pressure is equal to or greater than 800 psig and the reactor protection system is energized with APRM protection and RBM interlocks in service.

S. Reactor Vessel Pressure - Unless otherwise indicated, reactor vessel pressures listed in the Technical Specifications are those measured by the reactor vessel steam space detector.

T. Refueling Outage - Refueling outage is the period of time between the shutdown of the unit prior to a refueling and the startup of the plant subsequent to that refueling. For the purpose of designating frequency of testing and surveillance, a refueling outage shall mean a regularly scheduled refueling outage; however, where such outages occur within 8 months of the completion of the previous refueling outage, the required surveillance testing need not be performed until the next regularly scheduled outage.

U. Secondary Containment Integrity - Secondary containment integrity means that the reactor building is intact and the following conditions are met:

1. At least one door in each access opening is closed.

2. The standby gas treatment system is operable.

3. All reactor building automatic ventilation system isolation valves are operable or are secured in the isolated position.

V. Shutdown - The reactor is in a shutdown condition when the reactor mode switch is in the shutdown mode position and no core alterations are being performed. When the mode switch is placed in the shutdown position a reactor scram is initiated, power to the control rod drives is removed, and the reactor protection system trip systems are de-energized.

1. Hot Shutdown means conditions as above with reactor coolant temperature greater that 212°F.

2. Cold Shutdown means conditions as above with reactor coolant temperature equal to or less than 212°F.

3. Shutdown means conditions as above such that the effective multiplication factor (K_{eff}) of the core shall be less than 0.99.

W. Simulated Automatic Actuation - Simulated automatic actuation means applying a simulated signal to the sensor to actuate circuit in question.

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X. Transition Boiling - Transition boiling means the boiling regime between nucleate and film boiling. Transition boiling is the regime in which both nucleate and film boiling occur intermittently with neither type being completely stable.

Y. Surveillance Frequency - Unless otherwise stated in these specifications, periodic surveillance tests, checks, calibrations, and examinations shall be performed within the specified surveillance intervals. These intervals may be adjusted plus 25%. The operating cycle interval is considered to be 18 months and the tolerance stated above is applicable.

Z. Surveillance Interval - The surveillance interval is the calendar time between surveillance tests, checks, calibrations, and examinations to be performed upon an instrument or component when it is required to be operable. These tests unless otherwise stated in these specifications may be waived when the instrument, component, or system is not required to be operable, but these tests shall be performed on the instrument, component, or system prior to being required to be operable.

AA. Vital Fire Suppression Water System - The vital fire suppression water system is that part of the fire suppression system which protects those instruments, components, and systems required to perform a safe shutdown of the reactor. The vital fire suppression system includes the water supply, pumps, and distribution piping with associated sectionalizing valves, which provide immediate coverage of the Reactor Building, Control Room Building, and Diesel Generator Rooms.

BB. Source Check - The qualitative assessment of channel response when the channel sensor is exposed to a radioactive source.

CC. Dose Equivalent I-131 - The dose equivalent I-131 shall be that concentration of I-131 (microcurie/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in NRC Regulatory Guide 1.109, Revision 1, October 1977.

DD. Solidification - Solidification shall be the conversion of wet wastes into a form that meets shipping and burial ground requirements. Suitable forms include dewatered resins and filter sludges.

EE. Member(s) of the Public - Members of the public shall include all persons who are not occupationally associated with the plant. This category does not include employees of the utility, its contractors or vendors. Also excluded from this category are casual visitors to the plant and persons who enter the site to service equipment or to make deliveries.

FF. Site Boundary - The site boundary is shown in Figure 2.2-5 in the FSAR.

GG. Deleted

HH. Deleted

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II. Off-Site Dose Calculation Manual (ODCM) - A manual containing the current methodology and parameters used in the calculation of off-site doses due to radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints, and in the conduction of the environmental radiological monitoring program.

JJ. Process Control Program (PCP) - A process control program shall contain the sampling, analysis, tests, and determinations by which wet radioactive waste from liquid systems is assured to be converted to a form suitable for off-site disposal.

KK. Gaseous Radwaste Treatment System - The Augmented Off-Gas System (AOG) is the gaseous radwaste treatment system which has been designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system off-gases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

LL. Ventilation Exhaust Treatment System - The Radwaste Building and AOG Building ventilation HEPA filters are ventilation exhaust treatment systems which have been designed and installed to reduce radioactive material in particulate form in gaseous effluents by passing ventilation air through HEPA filters for the purpose of removing radioactive particulates from the gaseous exhaust stream prior to release to the environment. Engineered safety feature atmospheric cleanup systems, such as the Standby Gas Treatment (SBGT) System, are not considered to be ventilation exhaust treatment system components.

MM. Vent/Purging - Vent/Purging is the controlled process of discharging air or gas from the primary containment to control temperature, pressure, humidity, concentration or other operating conditions.

NN. Core Operating Limits Report - The Core Operating Limits Report is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.7.A.4. Plant operation within these operating limits is addressed in individual specifications.

1.1 SAFETY LIMIT

1.1 FUEL CLADDING INTEGRITY

Applicability:

Applies to the interrelated variable associated with fuel thermal behavior.

Objective:

To establish limits below which the integrity of the fuel cladding is preserved.

Specification:

A. Bundle Safety Limit (Reactor Pressure >800 psia and Core Flow >10% of Rated)

When the reactor pressure is >800 psia and the core flow is greater than 10% of rated, either:

1. For a core loading which consists of at least two successive reloads of P8x8R, BP8x8R, GE8x8E, or GE8x8EB fuel with high ($\geq$1.04) beginning-of-life bundle R-factor (one reload of which is fuel in its first cycle of operation), the existence of a Minimum Critical Power Ratio (MCPR) of less than 1.04 (1.05 for Single Loop Operation) shall constitute violation of the Fuel Cladding Integrity Safety Limit (FCISL);

or

2. For all other core loadings, the existence of a MCPR less than 1.07 (1.08 for Single Loop Operation) shall constitute violation of the FCISL.

2.1 LIMITING SAFETY SYSTEM SETTING

2.1 FUEL CLADDING INTEGRITY

Applicability:

Applies to trip setting of the instruments and devices which are provided to prevent the nuclear system safety limits from being exceeded.

Objective:

To define the level of the process variable at which automatic protective action is initiated.

Specification:

A. Trip Settings

The limiting safety system trip settings shall be as specified below:

1. Neutron Flux Trip Settings

a. APRM Flux Scram Trip Setting (Run Mode)

When the mode switch is in the RUN position, the APRM flux scram trip setting shall be as shown on Figure 2.1.1 and shall be:

$$S \leq 0.66(W-\Delta W)+54\%$$

where:

S = setting in percent of rated thermal power (1593 MWt)

W = percent rated two loop drive flow where 100% rated drive flow is that flow equivalent to 48 x 10^6 lbs/hr core flow

1.1 SAFETY LIMIT

B. Core Thermal Power Limit (Reactor Pressure $\leq$ 800 psia of Core Flow $\leq$ 10% or Rated)

When the reactor pressure is $\leq$800 psia or core flow $\leq$10% of rated, the core thermal power shall not exceed 25% of rated thermal power.

C. Power Transient

To ensure that the safety limit established in Specification 1.1A and 1.1B is not exceeded, each required scram shall be initiated by its expected scram signal. The safety limit shall be assumed to be exceeded when scram is accomplished by means other than the expected scram signal.

D. Whenever the reactor is shutdown with irradiated fuel in the reactor vessel, the water level shall not be less than 12 inches above the top of the enriched fuel when it is seated in the core.

2.1 LIMITING SAFETY SYSTEM SETTING

ΔW = difference between two loop and single loop drive flow at the same core flow. This difference must be accounted for during single loop operation. $\Delta W = 0$ for two loop operation.

In the event of operation with the ratio of MFLPD to FRP greater than 1.0, the APRM gain shall be increased by the ratio: $\frac{MFLPD}{FRP}$

where:

MFLPD = maximum fraction of limiting power density where the limiting power density is defined in the Core Operating Report.

FRP = fraction of rated power (1593 MWt).

In the event of operation with the ratio of MFLPD to FRP equal to or less than 1.0, the APRM gain shall be equal to or greater than 1.0.

1.1 SAFETY LIMIT

2.1 LIMITING SAFETY SYSTEM SETTING

For no combination of loop recirculation flow rate and core thermal power shall the APRM flux scram trip setting be allowed to exceed 120% of rated thermal power.

b. Flux Scram Trip Setting (Refuel or Startup and Hot Standby Mode)

When the reactor mode switch is in the REFUEL or STARTUP position, average power range monitor (APRM) scram shall be set down to less than or equal to 15% of rated neutron flux (except as allowed by Note 12 of Table 3.1.1). The IRM flux scram setting shall be set at less than or equal to 120/125 of full scale.

B. APRM Rod Block Trip Setting

1. The APRM rod block trip setting shall be as shown in Figure 2.1.1 and shall be:

$$S_{RB} \leq 0.66(W-\Delta W)+42\%$$

where:

S_{RB} = rod block setting in percent of rated thermal power (1593 MWt)

W = percent rated two loop drive flow where 100% rated drive flow is that flow equivalent to 48×10^6 lbs/hr core flow

1.1 SAFETY LIMIT

2.1 LIMITING SAFETY SYSTEM SETTING

ΔW = difference between two loop and single loop drive flow at the same core flow. This difference must be accounted for during single loop operation. $\Delta W = 0$ for two loop operation.

In the event of operation with the ratio of MFLPD to FRP greater than 1.0, the APRM gain shall be increased by the ratio:

$$\frac{MFLPD}{FRP}$$

where:

MFLPD = maximum fraction of limiting power density where the limiting power density is defined in the Core Operating Limits Report.

FRP = fraction of rated power (1593 MWt).

In the event of operation with the ratio of MFLPD to FRP equal to or less than 1.0, the APRM gain shall be equal to or greater than 1.0.

C. Reactor low water level scram setting shall be at least 127 inches above the top of the enriched fuel.

1.1 SAFETY LIMIT

2.1 LIMITING SAFETY SYSTEM SETTING

D. Reactor low-low water level Emergency Core Cooling System (ECCS) initiation shall be at least 82.5 inches above the top of the enriched fuel.

E. Turbine stop valve scram shall be less than or equal to 10% valve closure from full open.

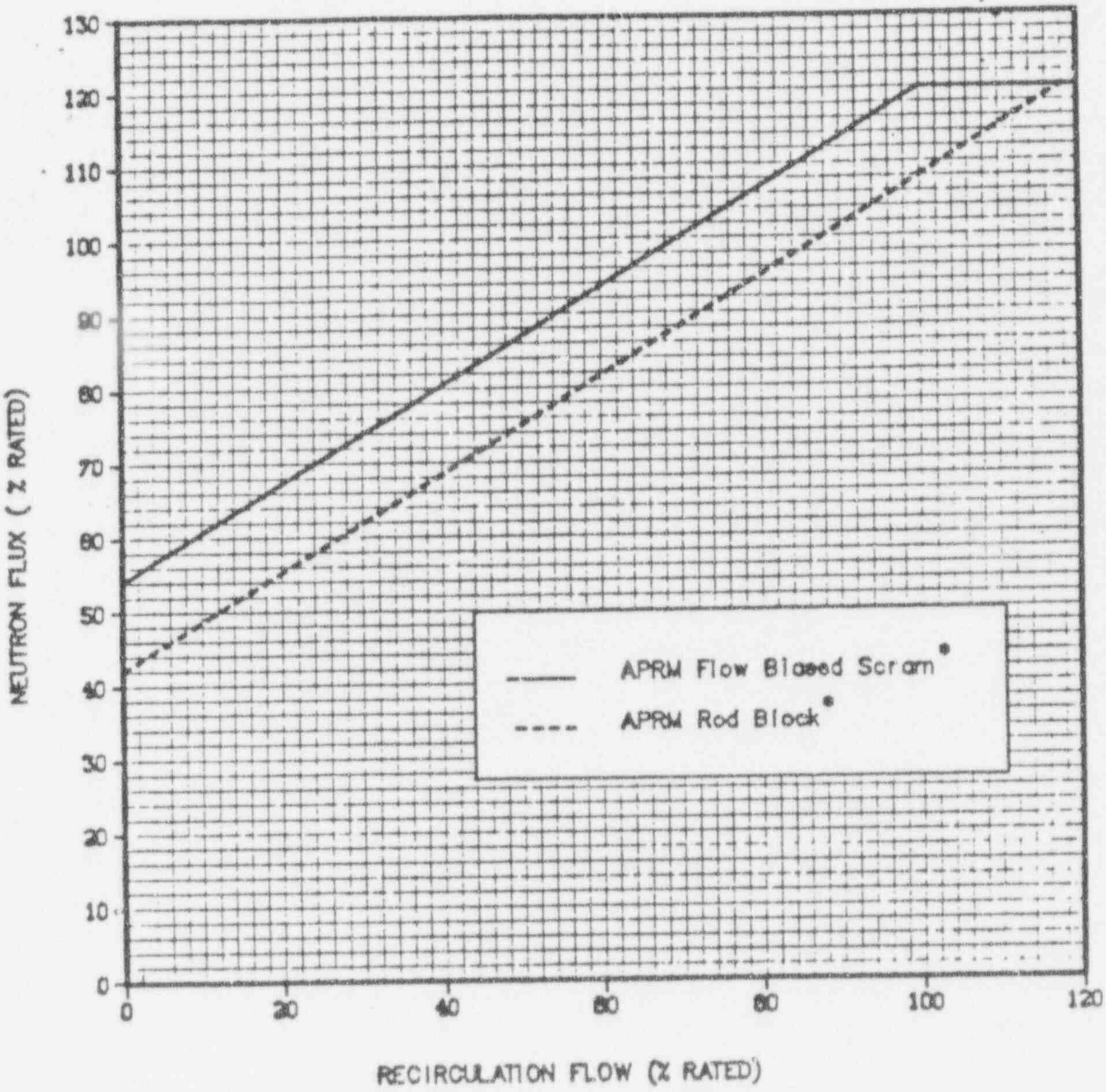
F. Turbine control valve fast closure scram shall, when operating at greater than 30% of full power, trip upon actuation of the turbine control valve fast closure relay.

G. Main steam line isolation valve closure scram shall be less than or equal to 10% valve closure from full open.

H. Main steam line low pressure initiation of main steam line isolation valve closure shall be at least 800 psig.

FIGURE 2.1-1

APRM FLOW REFERENCE SCRAM AND APRM ROD BLOCK SETTINGS



* For single loop operation, the APRM Scram and Rod Block settings are adjusted according to Technical Specifications 2.1.A.1.a and 2.1.B.1

BASES:

1.1 FUEL CLADDING INTEGRITY

Refer to Section S.2 of General Electric Company Licensing Topical Report, "United States Supplement, General Electric Standard Application for Reactor Fuel", NEDE-24011-P-A-US (Most Recent Revision).

The MCPR fuel cladding integrity safety limit is increased by 0.01 for single loop operation in order to account for increased core flow measurement and TIP reading uncertainties, as discussed in "Vermont Yankee Nuclear Power Station Single Loop Operation", NEDO-30060, February 1983.

B. Core Thermal Power Limit (Reactor Pressure ≤ 800 psia or Core Flow <10% of Rated)

At pressures below 800 psia, the core elevation pressure drop (0 power, 0 flow) is greater than 4.56 psi. At low power and all flows this pressure differential is maintained in the bypass region of the core. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and all flows will always be greater than 4.56 psi. Analyses show that with a flow of 28×10^3 lbs/hr bundle flow, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.56 psi driving head will be greater than 28×10^3 lbs/hr irrespective of total core flow and independent of bundle power for the range of bundle powers of concern. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors this corresponds to a core thermal power of more than 50%. Thus, a core thermal power limit of 25% for reactor pressures below 800 psia or core flow less than 10% is conservative.

C. Power Transient

Plant safety analyses have shown that the scrams caused by exceeding any safety setting will assure that the Safety Limit of Specification 1.1.1A or 1.1.1B will not be exceeded. Scram times are checked periodically to assure the insertion times are adequate. The thermal power transient resulting when a scram is accomplished other than by the expected scram signal (e.g., scram from neutron flux following closure of the main turbine stop valves) does not necessarily cause fuel damage. However, for this specification a Safety Limit violation will be assumed when a scram is only accomplished by means of a backup feature of the plant design. The concept of not approaching a Safety Limit provided scram signals are operable is supported by the extensive plant safety analysis.

The computer provided with Vermont Yankee has a sequence annunciation program which will indicate the sequence in which events such as scram, APRM trip initiation, pressure scram initiation, etc. occur. This program also indicates when the scram setpoint is cleared. This will provide information on how long a scram condition exists and thus provide some measure of the energy added during a transient.

BASES: 1.1 (Cont'd)

D. Reactor Water Level (Shutdown Condition)

During periods when the reactor is shutdown, consideration must also be given to water level requirements due to the effect of decay heat. If reactor water level should drop below the top of the enriched fuel during this time, the ability to cool the core is reduced. This reduction in core cooling capability could lead to elevated cladding temperatures and clad perforation. The core can be cooled sufficiently should the water level be reduced to two-thirds the core height. Establishment of the safety limit at 12 inches above the top of the enriched fuel provides adequate margin. This level will be continuously monitored.

BASES:

2.1 FUEL CLADDING INTEGRITY

A. Trip Settings

The bases for individual trip settings are discussed in the following paragraphs.

1. Neutron Flux Trip Settings

a. APRM Flux Scram Trip Setting (Run Mode)

The average power range monitoring (APRM) system, which is calibrated using heat balance data taken during steady state conditions, reads in percent of rated thermal power (1593 MWt). Because fission chambers provide the basic input signals, the APRM system responds directly to average neutron flux. During transients, the instantaneous rate of heat transfer from the fuel (reactor thermal power) is less than the instantaneous neutron flux due to the time constant of the fuel. Therefore, during abnormal operational transients, the thermal power of the fuel will be less than that indicated by the neutron flux at the scram setting. Analyses demonstrate that with a 120 percent scram trip setting, none of the abnormal operational transients analyzed violate the fuel Safety Limit and there is substantial margin from fuel damage. Therefore, the use of flow referenced scram trip provides even additional margin.

An increase in the APRM scram trip setting would decrease the margin present before the fuel cladding integrity Safety Limit is reached. The APRM scram trip setting was determined by an analysis of margins required to provide a reasonable range for maneuvering during operation. Reducing this operating margin would increase the frequency of spurious scrams which have an adverse effect on reactor safety because of the resulting thermal stresses. Thus, the APRM scram trip setting was selected because it provides adequate margin for the fuel cladding integrity Safety Limit yet allows operating margin that reduces the possibility of unnecessary scrams.

APRM Flux Scram Trip Setting (Run Mode)

The scram trip setting must be adjusted to ensure that the LHGR transient peak is not increased for any combination of MFLPD and reactor core thermal power. If the scram requires a change due to an abnormal peaking condition, it will be accomplished by increasing the APRM gain by the ratio in Specification 2.1.A.1.a, thus assuring a reactor scram at lower than design overpower conditions. For single recirculation loop operation, the APRM flux scram trip setting is reduced in accordance with the analysis presented in NEDO-30060, February 1983. This adjustment accounts for the difference between the single loop and two loop drive flow at the same core flow, and ensures that the margin of safety is not reduced during single loop operation.

Analyses of the limiting transients show that no scram adjustment is required to assure fuel cladding integrity when the transient is initiated from the operating limit MCPR defined in the Core Operating Limits Report.

BASES: 2.1 (Cont'd)

Flux Scram Trip Setting (Refuel or Startup and Hot Standby Mode)

For operation in the startup mode while the reactor is at low pressure, the reduced APRM scram setting to 15% of rated power provides adequate thermal margin between the setpoint and the safety limit, 25% of the rated. (During an outage when it is necessary to check refuel interlocks, the mode switch must be moved to the startup position. Since the APRM reduced scram may be inoperable at that time due to the disconnection of the LPRMs, it is required that the IRM scram and the SRM scram in noncoincidence be in effect. This will ensure that adequate thermal margin is maintained between the setpoint and the safety limit.) The margin is adequate to accommodate anticipated maneuvers associated with station startup. Effects of increasing pressure at zero or low void content are minor, cold water from sources available during startup is not much colder than that already in the system, temperature coefficients are small, and control rod patterns are constrained to be uniform by operating procedures backed up by the rod worth minimizer. Worth of individual rods is very low in a uniform rod pattern. Thus, of all possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power rise. Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks, and because several rods must be moved to change power by a significant percentage of rated power, the rate of power rise is very slow. Generally, the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the scram level, the rate of power rise is no more than 5% of rated power per minute, and the APRM system would be more than adequate to assure a scram before the power could exceed the safety limit. The reduced APRM scram remains active until the mode switch is placed in the RUN position. This switch can occur when reactor pressure is greater than 800 psig.

The IRM system consists of 6 chambers, 3 in each of the reactor protection system logic channels. The IRM is a 5-decade instrument, which covers the range of power level between that covered by the SRM and the APRM. The 5 decades are covered by the IRM by means of a range switch and the 5 decades are broken down into 10 ranges, each being one-half of a decade in size. The IRM scram trip setting of 120/125 of full scale is active in each range of the IRM. For example, if the instrument were on range 1, the scram setting would be a 120/125 of full scale for that range; likewise, if the instrument were on range 5, the scram would be 120/125 of full scale on that range. Thus, as the IRM is ranged up to accommodate the increase in power level, the scram trip setting is also ranged up. The most significant sources of reactivity change during the power increase are due to control rod withdrawal. For in-sequence control rod withdrawal, the rate of change of power is slow enough due to the physical limitation of withdrawing control rods, that heat flux is in equilibrium with the neutron flux and an IRM scram would result in a reactor shutdown well before any safety limit is exceeded.

BASES: 2.1 (Cont'd)

In order to ensure that the IRM provided adequate protection against the single rod withdrawal error, a range of rod withdrawal accidents was analyzed. This analysis included starting the accident at various power levels. The most severe case involves an initial condition in which the reactor is just subcritical and the IRM system is not yet on scale. This condition exists at quarter rod density. Additional conservatism was taken in this analysis by assuming that the IRM channel closest to the withdrawn rod is bypassed. The results of this analysis show that the reactor is scrammed and peak power limited to one percent of rated power, thus maintaining MCPR above the fuel cladding integrity safety limit. Based on the above analysis, the IRM provides protection against local control rod withdrawal errors and continuous withdrawal of control rods in sequence.

B. APRM Rod Block Trip Setting

Reactor power level may be varied by moving control rods or by varying the recirculation flow rate. The APRM system provides a control rod block to prevent rod withdrawal beyond a given point at the fuel cladding integrity safety limit. This rod block trip setting, which is automatically varied with recirculation loop flow rate, prevents an increase in the reactor power level to excessive values due to control rod withdrawal. The flow variable trip setting provides substantial margin from fuel damage, assuming a steady-state operation at the trip setting, over the entire recirculation flow range. The margin to the Safety Limit increases as the flow decreases for the specified trip setting versus flow relationship, therefore the worst case MCPR which could occur during steady-state operation is at 108% of rated thermal power because of the APRM rod block trip setting. The actual power distribution in the core is established by specified control rod sequences and is monitored continuously by the incore LPRM system. As with the APRM scram trip setting, the APRM rod block trip setting must be adjusted downward if the ratio of MFLPD to FRP exceeds the specified value. If the APRM rod block requires a change due to abnormal peaking conditions, it will be accomplished by increasing the APRM gain by the ratio in Specification 2.1B, thus ensuring a rod block at lower than design overpower conditions. As with the APRM flux scram trip setting, the APRM rod block trip setting is reduced for single recirculation loop operation in accordance with the analysis presented in NEDO-30060, February, 1983. This adjustment accounts for the difference between the single loop and two loop drive flow at the same core flow, and ensures that the margin of safety is not reduced during single loop operation.

C. Reactor Low Water Level Scram

The reactor low water level scram is set at a point which will prevent reactor operation with the steam separators uncovered, thus limiting carry-under to the recirculation loops. In addition, the safety limit is based on a water level below the scram point and therefore this setting is provided.

D. Reactor Low Water Level ECCS Initiation Trip Point

The core standby cooling subsystems are designed to provide sufficient cooling to the core to dissipate the energy associated with the loss-of-coolant accident and to limit fuel clad temperature to well below the clad melting temperature, and to limit clad

BASES: 2.1 (Cont'd)

metal-water reaction to less than 1%, to assure that core geometry remains intact.

The design of the ECCS components to meet the above criteria was dependent on three previously set parameters: the maximum break size, the low water level scram setpoint, and the ECCS initiation setpoint. To lower the ECCS initiation setpoint would now prevent the ECCS components from meeting their design criteria. To raise the ECCS initiation setpoint would be in a safe direction, but it would reduce the margin established to prevent actuation of the ECCS during normal operation or during normally expected transients.

E. Turbine Stop Valve Closure Scram Trip Setting

The turbine stop valve closure scram trip anticipates the pressure, neutron flux and heat flux increase that could result from rapid closure of the turbine stop valves. With a scram trip setting of <10% of valve closure from full open, the resultant increase in surface heat flux is limited such that MCPR remains above the fuel cladding integrity safety limit even during the worst case transient that assumes the turbine bypass is closed. This scram is bypassed when turbine steam flow is below 30% of rated, as measured by turbine first stage pressure.

F. Turbine Control Valve Fast Closure Scram

The control valve fast closure scram is provided to limit the rapid increase in pressure and neutron flux resulting from fast closure of the turbine control valves due to a load rejection coincident with failure of the bypass system. This transient is less severe than the turbine stop valve closure with failure of the bypass valves and therefore adequate margin exists.

G. Main Steam Line Isolation Valve Closure Scram

The isolation valve closure scram anticipates the pressure and flux transients which occur during normal or inadvertent isolation valve closure. With the scram setpoint at 10% of valve closure, there is no increase in neutron flux.

H. Reactor Coolant Low Pressure Initiation of Main Steam Isolation Valve Closure

The low pressure isolation of the main steam lines at 800 psig is provided to give protection against rapid reactor depressurization and the resulting rapid cooldown of the vessel. Advantage is taken of the scram feature which occurs when the main steam line isolation valves are closed, to provide the reactor shutdown so that high power operation at low reactor pressure does not occur. Operation of the reactor at pressures lower than 800 psig requires that the reactor mode switch be in the startup position where protection of the fuel cladding integrity safety limit is provided by the IRM high neutron flux scram.

Thus, the combination of main steam line low pressure isolation and isolation valve closure scram assures the available of neutron scram protection over the entire range of applicability of the fuel cladding integrity safety limit.

1.2 SAFETY LIMIT

1.2 REACTOR COOLANT SYSTEM

Applicability:

Applies to limits on reactor coolant system pressure.

Objective:

To establish a limit below which the integrity of the reactor coolant system is not threatened due to an overpressure condition.

Specification:

The reactor coolant system pressure shall not exceed 1335 psig at any time when irradiated fuel is present in the reactor vessel.

2.2 LIMITING SAFETY SYSTEM SETTING

2.2 REACTOR COOLANT SYSTEM

Applicability:

Applies to trip settings for controlling reactor system pressure.

Objective:

To provide for protective action in the event that the principle process variable approaches a safety limit.

Specification:

A. Reactor coolant high pressure scram shall be less than or equal to 1055 psig.

B. Primary system relief and safety valve settings shall be as follows:

1 valve at $\leq$1080 psig
2 valves at $\leq$1090 psig
1 valve at $\leq$1100 psig
2 valves at $\leq$1240 psig
(safety valves)

BASES:

1.2 REACTOR COOLANT SYSTEM

The reactor coolant system is an important barrier in the prevention of uncontrolled release of fission products. It is essential that the integrity of this system be protected by establishing a pressure limit to be observed for all operating conditions and whenever there is irradiated fuel in the reactor vessel.

The pressure safety limit of 1335 psig as measured by the vessel steam space pressure indicator is equivalent to 1375 psig at the lowest elevation of the reactor coolant system. The 1375 psig value is derived from the design pressures of the reactor pressure vessel, and the coolant system piping. The respective design pressures are 1250 psig at 575°F and 1148 psig at 560°F. The pressure safety limit was chosen as the lower of the pressure transients permitted by the applicable design codes: ASME Boiler and Pressure Vessel Code, Section III-A for the pressure vessel, ASME Boiler and Pressure Vessel Code Section III-C for the recirculation pump casing, and USASI B31.1 Code for the reactor coolant system piping. The ASME Boiler and Pressure Vessel Code permits pressure transients up to 10% over design pressure (110% x 1250 = 1375 psig), and the USASI Code permits pressure transients up to 20% over the design pressure (120% x 1148 = 1378 psig).

The safety valves are sized to prevent exceeding the pressure vessel code limit for the worst-case isolation (pressurization) event (MSIV closure) assuming indirect (neutron flux) scram.

2.2 REACTOR COOLANT SYSTEM

The settings on the reactor high pressure scram, reactor coolant system relief and safety valves, have been established to assure never reaching the reactor coolant system pressure safety limit as well as assuring the system pressure does not exceed the range of the fuel cladding integrity safety limit. In addition to preventing power operation above 1055 psig, the pressure scram backs up the APRM neutron flux scram for steam line isolation type transients. (See FSAR Section 14.5 and Supplement 2 to Proposed Change No. 14, November 12, 1973.)

3.1 LIMITING CONDITIONS FOR OPERATION

3.1 REACTOR PROTECTION SYSTEM

Applicability:

Applies to the operability of plant instrumentation and control systems required for reactor safety.

Objective:

To specify the limits imposed on plant operation by those instrument and control systems required for reactor safety.

Specification:

A. Plant operation at any power level shall be permitted in accordance with Table 3.1.1. The system response time from the opening of the sensor contact up to and including the opening of the scram solenoid relay shall not exceed 50 milliseconds.

B. During operation with the ratio of MFLPD to FRP greater than 1.0 either:

   a. The APRM System gains shall be adjusted by the ratios given in Technical Specifications 2.1.A.1 and 2.1.B or

   b. The power distribution shall be changed to reduce the ratio of MFLPD to FRP.

4.1 SURVEILLANCE REQUIREMENTS

4.1 REACTOR PROTECTION SYSTEM

Applicability:

Applies to the surveillance of the plant instrumentation and control systems required for reactor safety.

Objective:

To specify the type and frequency of surveillance to be applied to those instrument and control systems required for reactor safety.

Specification:

A. Instrumentation systems shall be functionally tested and calibrated as indicated in Tables 4.1.1 and 4.1.2, respectively.

B. Once a day during reactor power operation the maximum fraction of limiting power density and fraction of rated power shall be determined and the APRM system gains shall be adjusted by the ratios given in Technical Specifications 2.1.A.1.a and 2.1.B.

TABLE 3.1.1

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT REQUIREMENTS

	Trip Function	Trip Settings	Modes in Which Functions Must be Operating: Refuel (1)	Startup (12)	Run	Minimum Number Operating Instrument Channels Per Trip System (2)	Required Conditions When Minimum Conditions For Operation Are Not Satisfied (3)
1.	Mode Switch in Shutdown		X	X	X	1	A
2.	Manual Scram		X	X	X	1	A
3.	IRM						
	High Flux	≤120/125	X	X	X(11)	2	A
	INOP		X	X	X(11)	2	A
4.	APRM						
	High Flux (flow bias)	≤0.66 (W-ΔW)+54% (4)			X	2	A or B
	High Flux (reduced)	≤15%	X	X		2	A
	INOP				X	2(5)	A or B
	Downscale	≥2/125			X	2	A or B
5.	High Reactor Pressure	≤1055 psig	X	X	X	2	A
6.	High Drywell Pressure	≤2.5 psig	X	X	X	2	A
7.	Reactor Low (6) Water Level	≥127.0 inches	X	X	X	2	A
8.	Scram Discharge Volume High Level	≤21 gallons	X	X	X	2 (per volume)	A

TABLE 3.1.1
(Cont'd)

	Trip Function	Trip Settings	Modes in Which Functions Must be Operating: Refuel (1)	Startup (12)	Run	Minimum Number Operating Instrument Channels Per Trip System (2)	Required Conditions When Minimum Conditions For Operation Are Not Satisfied (3)
9.	Main steamline high radiation (7)	3x normal background at rated power(8)	X	X	X	2	A or C
10.	Main steamline isolation valve closure	≤10% valve closure			X	4	A or C
11.	Turbine control valve fast closure	(9)(10)			X	2	A or D
12.	Turbine stop valve closure	≤10% valve(10) closure			X	2	A or D

TABLE 3.1.1 NOTES

1. When the reactor is subcritical and the reactor water temperature is less than 212°F, only the following trip functions need to be operable:

   a) mode switch in shutdown
   b) manual scram
   c) high flux IRM or high flux SRM in coincidence
   d) scram discharge volume high water level

2. Whenever an instrument system is found to be inoperable, the instrument system output relay shall be tripped immediately. Except for MSIV and Turbine Stop Valve Position, this action shall result in tripping the trip system.

3. When the requirements in the column "Minimum Number of Operating Instrument Channels Per Trip System" cannot be met for one system, that system shall be tripped. If the requirements cannot be met for both trip systems, the appropriate actions listed below shall be taken:

   a) Initiate insertion of operable rods and complete insertion of all operable rods within four hours.
   b) Reduce power level to IRM range and place mode switch in the "Startup/Hot Standby" position within eight hours.
   c) Reduce turbine load and close main steam line isolation valves within 8 hours.
   d) Reduce reactor power to less than 30% of rated within 8 hours.

4. "W" is percent rated two loop drive flow where 100% rated drive flow is that flow equivalent to 48 x 10^6 lbs/hr core flow. ΔW is the difference between the two loop and single loop drive flow at the same core flow. This difference must be accounted for during single loop operation. $\Delta W = 0$ for two recirculation loop operation.

5. To be considered operable an APRM must have at least 2 LPRM inputs per level and at least a total of 13 LPRM inputs, except that channels A, C, D, and F may lose all LPRM inputs from the companion APRM Cabinet plus one additional LPRM input and still be considered operable.

6. The top of the enriched fuel has been designated as 0 inches and provides common reference level for all vessel water level instrumentation.

7. Channel shared by the Reactor Protection and Primary Containment Isolation Systems.

8. An alarm setting of 1.5 times normal background at rated power shall be established to alert the operator to abnormal radiation levels in primary coolant.

TABLE 3.1.1 NOTES (Cont'd)

9. Channel signals for the turbine control valve fast closure trip shall be derived from the same event or events which cause the control valve fast closure.

10. A turbine stop valve closure and generator load rejection bypass is permitted when the first stage turbine pressure is less than 30% of normal (220 psia).

11. The IRM scram is bypassed when the APRMs are on scale and the mode switch is in the run position.

12. While performing refuel interlock checks which require the mode switch to be in Startup, the reduced APRM high flux scram need not be operable provided:

    a. The following trip functions are operable:

        1. Mode switch in shutdown,
        2. Manual scram,
        3. High flux IRM scram
        4. High flux SRM scram in noncoincidence,
        5. Scram discharge volume high water level, and;

    b. No more than two (2) control rods withdrawn. The two (2) control rods that can be withdrawn cannot be faced adjacent or diagonally adjacent.

TABLE 4.1.1

SCRAM INSTRUMENTATION AND LOGIC SYSTEMS FUNCTIONAL TESTS

MINIMUM FUNCTIONAL TEST FREQUENCIES FOR SAFETY INSTRUMENTATION, LOGIC SYSTEMS AND CONTROL CIRCUITS

Instrument Channel	Group[3]	Functional Test[7]	Minimum Frequency[4]
Mode Switch in Shutdown	A	Place Mode Switch in Shutdown	Each Refueling Outage
Manual Scram	A	Trip Channel and Alarm	Every 3 Months
IRM			
High Flux	C	Trip Channel and Alarm[5]	Before Each Startup & Weekly During Refueling[6]
Inoperative	C	Trip Channel and Alarm	Before Each Startup & Weekly During Refueling[6]
APRM			
High Flux	B	Trip Output Relays[5]	Once Each Week
High Flux (Reduced)	B	Trip Output Relays[5]	Before Each Startup & Weekly During Refueling[6]
Inoperative	B	Trip Output Relays	Once Each Week
Downscale	B	Trip Output Relays[5]	Once Each Week
Flow Bias	B	Trip Output Relays[5]	(1)
High Reactor Pressure	B	Trip Channel and Alarm[5]	(1)
High Drywell Pressure	B	Trip Channel and Alarm[5]	(1)
Low Reactor Water Level[2][8]	B	Trip Channel and Alarm[5]	(1)
High Water Level in Scram Discharge Volumes	B	Trip Channel and Alarm[5]	(1)
High Main Steam Line Radiation[2]	B	Trip Channel and Alarm[5]	Once Each Week
Main Steam Line Iso. Valve Closure	A	Trip Channel and Alarm	(1)
Turbine Con. Valve Fast Closure	A	Trip Channel and Alarm	(1)
Turbine Stop Valve Closure	A	Trip Channel and Alarm	(1)
Scram Test Switch	A	Trip Channel and Alarm	Each Refueling Outage
First Stage Turbine Pressure - Permissive	A	Trip Channel and Alarm	Every 6 Months

TABLE 4.1.1 NOTES

1. Initially once per month, thereafter, with an interval not less than one month nor more than three months. The compilation of instrument failure rate data may include data obtained from other Boiling Water Reactors for which the same design instrument operates in an environment similar to that of Vermont Yankee.

2. An instrument check shall be performed on reactor water level and reactor pressure instrumentation once per day and on steamline radiation monitors once per shift.

3. A description of the three groups is included in the basis of this Specification.

4. Functional tests are not required when the systems are not required to be operable or are tripped. If tests are missed, they shall be performed prior to returning the systems to an operable status.

5. This instrumentation is exempted from the Instrument Functional Test Definition (1.G.). This Instrument Functional Test will consist of injecting a simulated electrical signal into the measurement channels.

6. Frequency need not exceed weekly.

7. A functional test of the logic of each channel is performed as indicated. This coupled with placing the mode switch in shutdown each refueling outage constitutes a logic system functional test of the scram system.

8. The water level in the reactor vessel will be perturbed and the corresponding level indicator changes will be monitored. This test will be performed every month after the completion of the monthly tests program.

TABLE 4.1.2

SCRAM INSTRUMENT CALIBRATION

MINIMUM CALIBRATION FREQUENCIES FOR REACTOR PROTECTION INSTRUMENT CHANNELS

Instrument Channel	Group(1)	Calibration Standard(4)	Minimum Frequency(2)
High Flux APRM			
Output Signal	B	Heat Balance	Once Every 7 Days
Output Signal (Reduced)	B	Heat Balance	Once Every 7 Days
Flow Bias	B	Standard Pressure and Voltage Source	Refueling Outage
LPRM	B(5)	Using TIP System	Every 1000 Equivalent Full Power Hours
High Reactor Pressure	B	Standard Pressure Source	Once/Operating Cycle
Turbine Control Valve Fast Closure	A	Standard Pressure Source	Every 3 Months
High Drywell Pressure	B	Standard Pressure Source	Once/Operating Cycle
High Water Level in Scram Discharge Volume	B	Water Level	Once/Operating Cycle
Low Reactor Water Level	B	Standard Pressure Source	Once/Operating Cycle
Turbine Stop Valve Closure	A	(6)	Refueling Outage
High Main Steam Line Radiation	B	Appropriate Radiation Source(3)	Refueling Outage
First Stage Turbine Pressure Permissive	A	Pressure Source	Every 6 Months and After Refueling
Main Steam Line Isolation Valve Closure	A	(6)	Refueling Outage

TABLE 4.1.2 NOTES

1. A description of the three groups is included in the bases of this Specification.

2. Calibration tests are not required when the systems are not required to be operable or are tripped. If tests are missed, they shall be performed prior to returning the systems to an operable status.

3. A current source provides an instrument channel alignment every 3 months.

4. Response time is not part of the routine instrument check and calibration, but will be checked every operating cycle.

5. Does not provide scram function.

6. Physical inspection and actuation.

BASES:

3.1 Reactor Protection System

The reactor protection system automatically initiates a reactor scram to:

1. preserve the integrity of the fuel barrier;
2. preserve the integrity of the primary system barrier; and
3. minimize the energy which must be absorbed, and prevent criticality following a loss of coolant accident.

This specification provides the limiting conditions for operation necessary to preserve the ability of the system to tolerate single failures and still perform its intended function even during periods when instrument channels may be out of service because of maintenance, testing, or calibration.

The reactor protection system is of the dual channel type. The system is made up of two independent logic channels, each having three subsystems of tripping devices. One of the three subsystems has inputs from the manual scram push buttons and the reactor mode switch. Each of the two remaining subsystems has an input from at least one independent sensor monitoring each of the critical parameters. The outputs of these subsystems are combined in a 1 out of 2 logic; i.e., an input signal on either one or both of the subsystems will cause a trip system trip. The outputs of the trip systems are arranged so that a trip on both logic channels is required to produce a reactor scram.

The required conditions when the minimum instrument logic conditions are not met are chosen so as to bring station operation promptly to such a condition that the particular protection instrument is not required; or the station is placed in the protection or safe condition that the instrument initiates. This is accomplished in a normal manner without subjecting the plant to abnormal operating conditions.

When the minimum requirements for the number of operable or operating trip system and instrumentation channels are satisfied, the effectiveness of the protection system is preserved; i.e., the system can tolerate a single failure and still perform its intended function of scramming the reactor.

Three APRM instrument channels are provided for each protection trip system to provide for high neutron flux protection. APRM's A and E operate contacts in a trip subsystem, and APRM's C an E operate contacts in the other trip subsystem. APRM's B, D, and F are arranged similarly in the other protection trip system. Each protection trip system has one more APRM than is necessary to meet the minimum number required. This allows the bypassing of one APRM per protection trip system for maintenance, testing, or calibration without changing the minimum number of channels required for inputs to each trip system. Additional IRM channels have also been provided to allow bypassing of one such channel. IRM assignment to the bypass switches is described on FSAR Figure 7.5-9 and on FSAR Page 7.5-8.

The bases for the scram settings for the IRM, APRM, high reactor pressure, reactor low water level, turbine control valve fast closure, and turbine stop valve closure are discussed in Specification 2.1.

Instrumentation (pressure switches) is provided to detect a loss-of-coolant accident and initiate the core standby cooling equipment. This instrumentation is a backup to the water level instrumentation which is discussed in Specification 3.2.

BASES: 3.1 (Cont'd)

The Control Rod Drive Scram System is designed so that all of the water that is discharged from the reactor by the scram can be accommodated in the discharge piping. This discharge piping is divided into two sections. One section services the control rod drives on the north side of the reactor, the other serves the control rod drives of the south side. A part of the piping in each section is an instrument volume which accommodates in excess of 21 gallons of water and is at the low point in the piping. No credit was taken for this volume in the design of the discharge piping as concerns the amount of water which must be accommodated during a scram. During normal operation, the discharge volume is empty; however, should it fill with water, the water discharged to the piping from the reactor could not be accommodated, which would result in slow scram times or partial or no control rod insertion. To preclude this occurrence, level instrumentation has been provided for the instrument volume which scram the reactor when the volume of water reaches 21 gallons. As indicated above, there is sufficient volume in the piping to accommodate the scram without impairment of the scram times or amount of insertion of the control rods. This function shuts the reactor down while sufficient volume remains to accommodate the discharged water, and precludes the situation in which a scram would be required but not be able to perform its function adequately. The present design of the Scram Discharge System is in concert with the BWR Owner's Group criteria, which have previously been endorsed by the NRC in their generic "Safety Evaluation Report (SER) for Scram Discharge Systems", dated December 1, 1980.

Loss of condenser vacuum occurs when the condenser can no longer handle the heat input. Loss of condenser vacuum initiates a closure of the turbine stop valves and turbine bypass valves which eliminates the heat input to the condenser. Closure of the turbine stop and bypass valves causes a pressure transient, neutron flux rise, and an increase in surface heat flux. To prevent the clad safety limit from being exceeded if this occurs, a reactor scram occurs on turbine stop valve closure. The turbine stop valve closure scram function alone is adequate to prevent the clad safety limit from being exceeded in the event of a turbine trip transient without bypass.

High radiation levels in the main steam line tunnel above that due to the normal nitrogen and oxygen radioactivity is an indication of leaking fuel. A scram is initiated whenever such radiation level exceeds three times normal background. The purpose of this scram is to reduce the source of such radiation to the extent necessary to prevent release of radioactive materials to the turbine. An alarm is initiated whenever the radiation level exceeds 1.5 times normal background to alert the operator to possible serious radioactivity spikes due to abnormal core behavior. The air ejector off-gas monitors serve to back up the main steam line monitors to provide further assurance against release of radioactive materials to site environs by isolating the main condenser off-gas line to the main stack.

The main steam line isolation valve closure scram is set to scram when the isolation valves are 10 percent closed from full open in 3-out-of-4 lines. This scram anticipates the pressure and flux transient, which would occur when the valves close. By scramming at this setting, the resultant transient is insignificant.

A reactor mode switch is provided which actuates or bypasses the various scram functions appropriate to the particular plant operating status.

The manual scram function is active in all modes, thus providing for manual means of rapidly inserting control rods during all modes of reactor operation.

BASES: 3.1 (Cont'd)

The IRM system provides protection against short reactor periods and, in conjunction with the reduced APRM system provides protection against excessive power levels in the startup and intermediate power ranges. A source range monitor (SRM) system is also provided to supply additional neutron level information during startup and can provide scram function with selected shorting links removed during refueling. Thus, the IRM and the reduced APRM are normally required in the startup mode and may be required in the refuel mode. During some refueling activities which require the mode switch in startup; it is allowable to disconnect the LPRMs to protect them from damage during under vessel work. In lieu of the protection provided by the reduced APRM scram, both the IRM scram and the SRM scram in noncoincidence are used to provide neutron monitoring protection against excessive power levels. In the power range, the normal APRM system provides required protection. Thus, the IRM system and 15% APRM scram are not required in the run mode. The requirement that the IRMs be inserted in the core until the APRMs read at least 2/125 of full scale assures that there is proper overlap in the neutron monitoring systems.

If an unsafe failure is detected during surveillance testing, it is desirable to determine as soon as possible if other failures of a similar type have occurred and whether the particular function involved is still operable or capable of meeting the single failure criteria. To meet the requirements of Table 3.1.1, it is necessary that all instrument channels in one trip system be operable to permit testing in the other trip system.

Thus, when failures are detected in the first trip system tested, they would have to be repaired before testing of the other system could begin. In the majority of cases, repairs or replacement can be accomplished quickly. If repair or replacement cannot be completed in a reasonable time, operation could continue with one tripped system until the surveillance testing deadline.

The requirement to have all scram functions, except those listed in Table 3.1.1, operable in the "Refuel" mode is to assure that shifting to this mode during reactor operation does not diminish the need for the reactor protection system.

The ability to bypass one instrument channel when necessary to complete surveillance testing will preclude continued operation with scram functions which may be either unable to meet the single failure criteria or completely inoperable. It also eliminates the need for an unnecessary shutdown if the remaining channels and subsystems are found to be operable. The conditions under which the bypass is permitted require an immediate determination that the particular function is operable. However, during the time a bypass is applied, the function will not meet the single failure criteria; therefore, it is prudent to limit the time the bypass is in effect by requiring that surveillance testing proceed on a continuous basis and that the bypass be removed as soon as testing is completed.

Sluggish indicator response during the perturbation test will be indicative of a plugged instrument line or closed instrument valves. Testing immediately after functional testing will assure the operability of the instrument lines. This test assures the operability of the reactor pressure sensors as well as the reactor level sensors since both parameters are monitored through the same instrument lines.

<u>BASES</u>: 3.1 (Cont'd)

The independence of the safety system circuitry is determined by operation of the scram test switch. Operation of this switch during the refueling outage and following maintenance on these circuits will assure their continued independence.

The calibration frequency, using the TIP system, specified for the LPRMs will provide assurance that the LPRM input to the APRM system will be corrected on a timely basis for LPRM detector depletion characteristics.

BASES:

4.1 REACTOR PROTECTION SYSTEM

A. The scram sensor channels listed in Tables 4.1.1 and 4.1.2 are divided into three groups: A, B, and C. Sensors that make up Group A are the on-off type and will be tested and calibrated at the indicated intervals. Initially the tests are more frequent than Yankee experience indicates necessary. However, by testing more frequently, the confidence level with this instrumentation will increase and testing will provide data to justify extending the test intervals as experience is accrued.

Group B devices utilize an analog sensor followed by an amplifier an bi-stable trip circuit. This type of equipment incorporates control room mounted indicators and annunciator alarms. A failure in the sensor or amplifier may be detected by an alarm or by an operator who observes that one indicator does not track tho others in similar channels. The bistable trip circuit failures are detected by the periodic testing.

Group C devices are active only during a given portion of the operating cycle. For example, the IRM is active during start-up and inactive during full-power operation. Testing of these instruments is only meaningful within a reasonable period prior to their use.

B. The ratio of MFLPD to FRP shall be checked once per day to determine if the APRM gains require adjustment. Because few control rod movements or power changes occur, checking these parameters daily is adequate.

3.2 LIMITING CONDITIONS FOR OPERATION

3.2 PROTECTIVE INSTRUMENT SYSTEMS

Applicability:

Applies to the operational status of the plant instrumentation systems which initiate and control a protective function.

Objective:

To assure the operability of protective instrumentation systems.

Specification:

A. Emergency Core Cooling System

When the system(s) it initiates or controls is required in accordance with Specification 3.5, the instrumentation which initiates the emergency core cooling system(s) shall be operable in accordance with Table 3.2.1.

B. Primary Containment Isolation

When primary containment integrity is required, in accordance with Specification 3.7, the instrumentation that initiates primary containment isolation shall be operable in accordance with Table 3.2.2.

C. Reactor Building Ventilation Isolation and Standby Gas Treatment System Initiation

The instrumentation that initiates the isolation of the reactor building ventilation system and the actuation of the standby gas treatment system shall be operable in accordance with Table 3.2.3.

4.2 SURVEILLANCE REQUIREMENTS

4.2 PROTECTIVE INSTRUMENT SYSTEMS

Applicability:

Applies to the surveillance requirements of the instrumentation systems which initiate and control a protective function.

Objective:

To verify the operability of protective instrumentation systems.

Specification:

A. Emergency Core Cooling System

Instrumentation and logic systems shall be functionally tested and calibrated as indicated in Table 4.2.1.

B. Primary Containment Isolation

Instrumentation and logic systems shall be functionally tested and calibrated as indicated in Table 4.2.2.

C. Reactor Building Ventilation Isolation and Standby Gas Treatment System Initiation

Instrumentation and logic systems shall be functionally tested and calibrated as indicated in Table 4.2.3.

3.2 LIMITING CONDITIONS FOR OPERATION

D. Off-Gas System Isolation

During reactor power operation, the instrumentation that initiates isolation of the off-gas system shall be operable in accordance with Table 3.2.4.

E. Control Rod Block Actuation

During reactor power operation the instrumentation that initiates control rod block shall be operable in accordance with Table 3.2.5.

F. Mechanical Vacuum Pump Isolation

1. Whenever the main steam line isolation valves are open, the mechanical vacuum pump shall be capable of being automatically isolated and secured by a signal of high radiation in the main steam line tunnel or shall be manually isolated and secured.

2. If Specification 3.2.F.1 is not met following a routine surveillance check, the reactor shall be in the cold shutdown within 24 hours.

G. Post-Accident Instrumentation

During reactor power operation, the instrumentation that displays information in the Control Room necessary for the operator to initiate and control the systems used during and following a postulated accident of abnormal operating condition shall be operable in accordance with Table 3.2.6.

4.2 SURVEILLANCE REQUIREMENTS

D. Off-Gas System Isolation

Instrumentation and logic systems shall be functionally tested and calibrated as indicated in Table 4.2.4.

E. Control Rod Block Actuation

Instrumentation and logic systems shall be functionally tested and calibrated as indicated in Table 4.2.5.

F. Mechanical Vacuum Pump Isolation

During each operating cycle, automatic isolation and securing of the mechanical vacuum pump shall be verified while the reactor is shutdown.

G. Post-Accident Instrumentation

The post-accident instrumentation shall be functionally tested and calibrated in accordance with Table 4.2.6.

3.2 LIMITING CONDITIONS FOR OPERATION

H. Drywell to Torus ΔP Instrumentation

1. During reactor power operation, the Drywell to Torus ΔP Instrumentation (recorder #1-156-3 and instrument DPI-1-158-6) shall be operable except as specified in 3.2.H.2.

2. From and after the date that one of the Drywell to Torus ΔP instruments is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding thirty days unless the instrument is sooner made operable. If both instruments are made or found to be inoperable, and indication cannot be restored within a six hour period, an orderly shutdown shall be initiated and the reactor shall be in a hot shutdown condition in six hours and a cold shutdown condition in the following eighteen hours.

I. Recirculation Pump Trip Instrumentation

During reactor power operation, the Recirculation Pump Trip Instrumentation shall be operative in accordance with Table 3.2.1.

J. Deleted

K. Degraded Grid Protective System

During reactor power operation, the emergency bus undervoltage instrumentation shall be operative in accordance with Table 3.2.8.

4.2 SURVEILLANCE REQUIREMENTS

H. Drywell to Torus ΔP Instrumentation

The Drywell to Torus ΔP Instrumentation shall be calibrated once every six months and an instrument check will be made once per shift.

I. Recirculation Pump Trip Instrumentation

The Recirculation Pump Trip Instrumentation shall be functionally tested and calibrated in accordance with Table 4.2.1.

J. Deleted

K. Degraded Grid Protective System

The emergency bus undervoltage instrumentation shall be functionally tested and calibrated in accordance with Table 4.2.8.

3.2 LIMITING CONDITIONS FOR OPERATION

L. Reactor Core Isolation Cooling System Actuation

When the Reactor Core Isolation Cooling System is required in accordance with Specification 3.5.G, the instrumentation which initiates actuation of this system shall be operable in accordance with Table 3.2.9.

4.2 SURVEILLANCE REQUIREMENTS

L. Reactor Core Isolation Cooling System Actuation

Instrumentation and Logic Systems shall be functionally tested and calibrated as indicated in Table 4.2.9.

TABLE 3.2.1

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

Core Spray - A & B (Note 1)

Minimum Number of Operable Instrument Channels per Trip System	Trip Function	Trip Level Setting	Required Action When Minimum Conditions for Operation Are Not Satisfied
2	High Drywell Pressure	$\leq$2.5 psig	Note 2
2	Low-Low Reactor Vessel Water Level	$\geq$82.5" above top of enriched fuel	Note 2
1	Low Reactor Pressure (PT-2-3-56C/D(S1)	300 $\leq$ P $\leq$ 350 psig	Note 2
2	Low Reactor Pressure (PT-2-3-56A/B(S1) & 52C/D(M))	300 $\leq$ P $\leq$ 350 psig	Note 2
1	Time Delay (14A-K16A & B)	$\leq$10 seconds	Note 2
2	Pump 14-1A, Discharge Pressure	$\geq$100 psig	Note 5
1	Auxiliary Power Monitor	--	Note 5
1	Pump Bus Power Monitor	--	Note 5
1	High Sparger Pressure	$\leq$5 psid	Note 5
1	Trip System Logic	--	Note 5

TABLE 3.2.1
(Cont'd)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

Low Pressure Coolant Injection System A & B (Note 1)

Minimum Number of Operable Instrument Channels per Trip System	Trip Function	Trip Level Setting	Required Action When Minimum Conditions for Operation Are Not Satisfied
1	Low Reactor Pressure (PT-2-3-56C/D(S1))	300 $\leq$ p $\leq$ 350 psig	Note 2
2	High Drywell Pressure (PT-10-101A-D(M))	$\leq$2.5 psig	Note 2
2	Low-Low Reactor Vessel Water Level	$\geq$82.5" above top of enriched fuel	Note 2
1	Time Delay (10A-K51A & B)	0 seconds	Note 5
1	Reactor Vessel Shroud Level	$\geq$2/3 core height	Note 5
1	Time Delay (10A-K72A & B)	$\leq$60 seconds	Note 5
1	Time Delay (10A-K50A & B)	$\leq$5 seconds	Note 5
1	Low Reactor Pressure (PS-2-128A & B)	100 $\leq$ p $\leq$ 150 psig	Note 2
2 per pump	RHR Pump A & C Discharge Pressure	$\geq$100 psig	Note 5
2	High Drywell Pressure (PT-10-101A-D(S1))	$\leq$2.5 psig	Note 2

TABLE 3.2.1
(Cont'd)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

Low Pressure Coolant Injection System A & B (Note 1)

Minimum Number of Operable Instrument Channels per Trip System	Trip Function	Trip Level Setting	Required Action When Minimum Conditions for Operation Are Not Satisfied
1	Time Delay (10A-K45A & B)	$\leq$6 minutes	Note 5
2	Low Reactor Pressure (PT-2-3-56A/B(S1) & 52C/D(M))	300 $\leq$ p $\leq$ 350 psig	Note 2
1	Auxiliary Power Monitor	--	Note 5
1	Pump Bus Power Monitor	--	Note 5
1	Trip System Logic	--	Note 5

TABLE 3.2.1
(Cont'd)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

High Pressure Coolant Injection System

Minimum Number of Operable Instrument Channels per Trip System	Trip Function	Trip Level Setting	Required Action When Minimum Conditions for Operation Are Not Satisfied
2 (Note 3)	Low-Low Reactor Vessel Water Level	Same as LPCI	Note 5
2 (Note 4)	Low Condensate Storage Tank Water Level	≥ 3%	Note 5
2 (Note 3)	High Drywell Pressure	Same as LPCI	Note 5
1 (Note 3)	Bus Power Monitor	--	Note 5
1 (Note 4)	Trip System Logic	--	Note 5
2 (Note 7)	High Reactor Vessel Water Level	≤177 inches above top of enriched fuel	Note 5

TABLE 3.2.1
(Cont'd)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

Automatic Depressurization

Minimum Number of Operable Instrument Channels per Trip System (Note 4)	Trip Function	Trip Level Setting	Required Action When Minimum Conditions for Operation Are Not Satisfied
2	Low-Low Reactor Vessel Water Level	Same as Core Spray	Note 6
2	High Drywell Pressure	$\leq$2.5 psig	Note 6
1	Time Delay (2E-K5A and B)	$\leq$120 seconds	Note 6
1	Bus Power Monitor	--	Note 6
1	Trip System Logic	--	Note 6
2	Time Delay (2E-K16A and B, 2E-K17A and B)	$\leq$8 minutes	Note 6

TABLE 3.2.1
(Cont'd)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

Recirculation Pump Trip - A & B (Note 1)

Minimum Number of Operable Instrument Channels per Trip System	Trip Function	Trip Level Setting	Required Action When Minimum Conditions for Operation Are Not Satisfied
2	Low-Low Reactor Vessel Water Level	$\geq$ 6' 10.5" above top of enriched fuel	Note 2
2	High Reactor Pressure	$\leq$ 1150 psig	Note 2
2	Time Delays	$\leq$ 10 seconds	Note 2
1	Trip System Logic	--	Note 2

TABLE 3.2.1 NOTES

1. Each of the two Core Spray, LPCI and RPT, subsystems are initiated and controlled by a trip system. The subsystem "B" is identical to the subsystem "A".

2. If the minimum number of operable instrument channels are not available, the inoperable channel shall be tripped using test jacks or other permanently installed circuits. If the channel cannot be tripped by the means stated above, that channel shall be made operable within 24 hours or an orderly shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours.

3. One trip system with initiating instrumentation arranged in a one-out-of-two taken twice logic.

4. One trip system with initiating instrumentation arranged in a one-out-of-two logic.

5. If the minimum number of operable channels are not available, the system is considered inoperable and the requirements of Specification 3.5 apply.

6. Any one of the two trip systems will initiate ADS. If the minimum number of operable channels in one trip system is not available, the requirements of Specification 3.5.F.2 and 3.5.F.3 shall apply. If the minimum number of operable channels is not available in both trip systems, Specifications 3.5.F.3 shall apply.

7. One trip system arranged in a two-out-of-two logic.

TABLE 3.2.2

PRIMARY CONTAINMENT ISOLATION INSTRUMENTATION

Minimum Number of Operable Instrument Channels per Trip System	Trip Function	Trip Setting	Required Action When Minimum Conditions for Operation Are Not Satisfied (Note 2)
2	Low-Low Reactor Vessel Water Level	$\geq$82.5" above the top of enriched fuel	A
2 of 4 in each of 2 channels	High Main Steam Line Area Temperature	$\leq$212°F	B
2/steam line	High Main Steam Line Flow	$\leq$140% of rated flow	B
2/(Note 1)	Low Main Steam Line Pressure	$\geq$800 psig	B
2/(Note 6)	High Main Steam Line Flow	$\leq$40% of rated flow	B
2	Low Reactor Vessel Water Level	Same as Reactor Protection System	A
2	High Main Steam Line Radiation (7) (8)	$\leq$3 x background at rated power (9)	B
2	High Drywell Pressure	Same as Reactor Protection System	A
2/(Note 10)	Condenser Low Vacuum	$\leq$12" Hg absolute	A
1	Trip System Logic	--	A

TABLE 3.2.2
(Cont'd)

HIGH PRESSURE COOLANT INJECTION SYSTEM ISOLATION INSTRUMENTATION

Minimum Number of Operable Instrument Channels per Trip System	Trip Function	Trip Level Setting	Required Action When Minimum Conditions for Operation Are Not Satisfied
2 per set of 4	High Steam Line Space Temperature	≤212°F	Note 3
1	High Steam Line d/p (Steam Line Break)	≤195 inches of water	Note 3
4 (Note 5)	Low HPCI Steam Supply Pressure	≥70 psig	Note 3
2	Main Steam Line Tunnel Temperature	≤212°F	Note 3
1	Time Delay (23A-K48) (23A-K49)	≤35 minutes	Note 3
1	Bus Power Monitor	--	--
1	Trip System Logic	--	--

TABLE 3.2.2
(Cont'd)

REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION INSTRUMENTATION

Minimum Number of Operable Instrument Channels per Trip System	Trip Function	Trip Level Setting	Required Action When Minimum Conditions for Operation Are Not Satisfied (Note 2)
2	Main Steam Line Tunnel Temperature	$\leq$212°F	Note 3
1	Time Delay (13A-K41) (13A-K42)	$\leq$35 minutes	Note 3
2 per set of 4	High Steam Line Space Temperature	$\leq$212°F	Note 3
1	High Steam Line d/p (Steam Line Break)	$\leq$195 inches of water	Note 3
4 (Note 5)	Low Steam Supply Pressure	$\geq$50 psig	Note 3
1	Bus Power Monitor	--	Note 3
1	Trip System Logic	--	Note 3
1	Time Delay (13A-K7) (13A-K31)	$3\leq t \leq 7$ seconds	Note 3

TABLE 3.2.2 NOTES

1. The main steam line low pressure need be available only in the "Run" mode.

2. If the minimum number of operable instrument channels are not available for one trip system, that trip system shall be tripped. If the minimum number of operable instrument channels are not available for both trip systems, the appropriate actions listed below shall be taken:

   A. Initiate an orderly shutdown and have reactor in the cold shutdown condition in 24 hours.

   B. Initiate an orderly load reduction and have reactor in "Hot Standby" within 8 hours.

3. Close isolation valves in system and comply with Specification 3.5.

4. Deleted.

5. One trip system arranged in a one-out-of-two twice logic.

6. The main steam line high flow is available only in the "Refuel," "Shutdown," and "Startup" modes.

7. This signal also automatically closes the mechanical vacuum pump suction line isolation valves.

8. Channel shared by the Reactor Protection and Primary Containment Isolation Systems.

9. An alarm setting of 1.5 times normal background at rated power shall be established to alert the operator to abnormal radiation levels in the primary coolant.

10. A key lock switch is provided to permit the bypass of this trip function to enable plant startup and shutdown when the condenser vacuum is greater than 12 inches Hg absolute provided that both turbine stop and bypass valves are closed.

TABLE 3.2.3

REACTOR BUILDING VENTILATION ISOLATION & STANDBY GAS TREATMENT SYSTEM INITIATION

Minimum Number of Operable Instrument Channels per Trip System	Trip Function	Trip Level Setting	Required Action When Minimum Conditions for Operation Are Not Met
2	Low Reactor Vessel Water Level	Same as PCIS	Note 1
2	High Drywell Pressure	Same as PCIS	Note 1
1	Reactor Building Vent	$\leq$14 mr/hr	Note 1
1	Refueling Floor Zone Radiation	$\leq$100 mr/hr	Note 1
1	Reactor Building Vent Trip System Logic	--	Note 1
1	Standby Gas Treatment Trip System Logic	--	Note 1
1	Logic Bus Power Monitor	--	Note 1

Note 1 - If the minimum number of operable instrument channels is not available in either trip system for more than 24 hours, the reactor building ventilation system shall be isolated and the standby gas treatment system operated until the instrumentation is repaired.

TABLE 3.2.4

OFF-GAS SYSTEM ISOLATION INSTRUMENTATION

Minimum Number of Operable Instrument Channels per Trip System	Trip Function	Trip Level Setting	Required Action When Minimum Conditions for Operation Are Not Met
1	Time Delay (Stack Off-Gas Valve Isolation) (15TD & 16TD)	$\leq$ 2 minutes $\leq$ 30 minutes	Note 1
1	Trip System Logic	--	Note 1

Note 1 - At least one of the radiation monitors between the charcoal bed system and the plant stack shall be operable during operation of the augmented off-gas system. If this condition cannot be met, continued operation of the augmented off-gas system is permissible for a period of up to 7 days provided that at least one of the stack monitoring systems is operable and off-gas system temperature and pressure are measured continuously.

TABLE 3.2.5

CONTROL ROD BLOCK INSTRUMENTATION

Minimum Number of Operable Instrument Channels per Trip System		Trip Function	Modes in Which Function Must be Operable: Refuel	Startup	Run	Trip Setting
		Startup Range Monitor				
(Note 1)	2	a. Upscale (Note 2)	X	X		$\leq 5 \times 10^5$ cps (Note 3)
	2	b. Detector Not Fully Inserted	X	X		
		Intermediate Range Monitor				
	2	a. Upscale	X	X		$\leq$108/125 Full Scale
	2	b. Downscale (Note 4)	X	X		$\geq$5/125 Full Scale
	2	c. Detector Not Fully Inserted	X	X		
		Average Power Range Monitor				
	2	a. Upscale (Flow Bias)			X	$\leq 0.66(W-\Delta W)+42\%$ (Note 5)
	2	b. Downscale			X	$\geq$2/125 Full Scale
		Rod Block Monitor (Note 6)				
(Note 9)	1	a. Upscale (Flow Bias) (Note 7)			X	$\leq 0.66(W-\Delta W)+N$ (Note 5)
	1	b. Downscale (Note 7)			X	$\geq$2/125 Full Scale
(Note 8)	1 (per volume)	Scram Discharge Volume	X	X	X	$\leq$12 Gallons
	1	Trip System Logic	X	X	X	

TABLE 3.2.5 NOTES

1. There shall be two operable or tripped trip systems for each function in the required operating mode. If the minimum number of operable instruments are not available for one of the two trip systems, this condition may exist for up to seven days provided that during the time the operable system is functionally tested immediately and daily thereafter; if the condition lasts longer than seven days, the system shall be tripped. If the minimum number of instrument channels are not available for both trip systems, the systems shall be tripped.

2. One of these trips may be bypassed. The SRM function may be bypassed in the higher IRM ranges when the IRM upscale rod block is operable.

3. This function may be bypassed when count rate is $\geq$100 cps or when all IRM range switches are above Position 2.

4. IRM downscale may be bypassed when it is on its lowest scale.

5. "W" is percent rated two loop drive flow where 100% rated drive flow is that flow equivalent to 48 x 10^6 lbs/hr core flow. Refer to the Core Operating Limits Report for acceptable values for N. ΔW is the difference between the two loop and single loop drive flow at the same core flow. This difference must be accounted for during single loop operation. $\Delta W = 0$ for two recirculation loop operation.

6. The minimum number of operable instrument channels may be reduced by one for maintenance and/or testing for periods not in excess of 24 hours in any 30-day period.

7. The trip may be bypassed when the reactor power is $\leq$30% of rated. An RBM channel will be considered inoperable if there are less than half the total number of normal inputs from any LPRM level.

8. With the number of operable channels less than required by the minimum operable channels per trip function requirement, place the inoperable channel in the tripped condition within one hour.

9. With one RBM channel inoperable:

   a. Verify that the reactor is not operating on a limiting control rod pattern, and

   b. Restore the inoperable RBM channel to operable status within 24 hours.

   Otherwise, place the inoperable rod block monitor channel in the tripped condition within the next hour.

TABLE 3.2.6

POST-ACCIDENT INSTRUMENTATION

Minimum Number of Operable Instrument Channels	Parameter	Type of Indication	Instrument Range
2	Drywell Atmospheric Temperature (Note 1)	Recorder #TR-16-19-45 (TE-16-19-30A) Meter #TI-16-19-30B	0-350°F 0-350°F
2	Containment Pressure (Note 1)	Meter #PI-16-19-12A Meter #PI-16-19-12B	0-275 psia 0-275 psia
2	Torus Pressure (Note 1)	Meter #PI-16-19-36A Meter #PI-16-19-36B	0-80 psia 0-80 psia
2	Torus Water Level (Note 3)	Meter #LI-16-19-12A Meter #LI-16-19-12B	0-25 ft. 0-25 ft.
2	Torus Water Temperature (Note 1)	Meter #16-19-33A Meter #16-19-33C	0-250°F 0-250°F
2	Reactor Pressure (Note 1)	Meter #PI-2-3-56A Meter #PI-2-3-56B	0-1500 psig 0-1500 psig
2	Reactor Vessel Water Level (Note 1)	Meter #2-3-91A Meter #2-3-91B	(-200)-0-(+200)" H_2O (-200)-0-(+200)" H_2O
1	Control Rod Position (Notes 1 and 2)	Meter	0-48" RPIS
1	Neutron Monitor (Notes 1 and 2)	Meter	0-125% Rated Flux
2	Torus Air Temperature (Note 1)	Recorder #TR-16-19-45 (TE-16-19-34) Meter #TI-16-19-41	0-350°F 50-300°F
2/valve	Safety/Relief Valve Position From Pressure Switches (Note 4)	Lights (SRV 2-71-A thru D)	Closed - Open

TABLE 3.2.6
(Cont'd)

POST-ACCIDENT INSTRUMENTATION

Minimum Number of Operable Instrument Channels	Parameter	Type of Indication	Instrument Range
1/valve	Safety Valve Position From Acoustic Monitor (Note 5)	Meter Z1-2-1A/B	Closed - Open
2	Containment Hydrogen/Oxygen Monitor (Note 1)	Meter SR-VG-6A Meter SR-VG-6B	0-30% hydrogen 0-25% oxygen
2	Containment High-Range Radiation Monitor (Note 6)	Meter RM-16-19-1A/B	1 R/hr-10^7 R/hr
1	Stack Noble Gas Effluent (Note 7)	Meter RM-17-155	0.1 - 10^7 mR/hr

TABLE 3.2.6 NOTES

Note 1 - From and after the date that a parameter is reduced to one indication, operation is permissible for 30 days. If a parameter is not indicated in the Control Room, continued operation is permissible during the next seven days. If indication cannot be restored within the next six hours, an orderly shutdown shall be initiated and the reactor shall be in a hot shutdown condition in six hours and a cold shutdown condition in the following 18 hours.

Note 2 - Control rod position and neutron monitor instruments are considered to be redundant to each other.

Note 3 - From and after the date that this parameter is reduced to one indication in the Control Room, continued reactor operation is permissible during the next 30 days. If both channels are inoperable and indication cannot be restored in six hours, an orderly shutdown shall be initiated and the reactor shall be in a hot shutdown condition in six hours and a cold shutdown condition in the following 18 hours.

Note 4 - From and after the date that safety/relief valve position from pressure switches is unavailable, reactor operation may continue provided safety/relief valve position can be determined from Recorder #2-166 (steam temperature in SRVs, 0-600°F) and Meter 16-19-33A or C (torus water temperature, 0-250°F). If both parameters are not available, the reactor shall be in a hot shutdown condition in six hours and a cold shutdown condition in the following 18 hours.

Note 5 - From and after the date that safety valve position from the acoustic monitor is unavailable, reactor operation may continue provided safety valve position can be determined from Recorder #2-166 (thermocouple, 0-600°F) and Meter #16-19-12A or B (containment pressure 0-275 psia). If both indications are not available, the reactor shall be in a hot shutdown condition in six hours and in a cold shutdown condition in the following 18 hours.

Note 6 - Within 30 days following the loss of one indication, or seven days following the loss of both indications, restore the inoperable channel(s) to an operable status or a special report to the Commission pursuant to Specification 6.7 must be prepared and submitted within the subsequent 14 days, outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the system to operable status.

Note 7 - From and after the date that this parameter is unavailable by Control Room indication, and cannot be restored within 24 hours, continued reactor operation is permissible for the next 30 days provided that local sampling capacity is available. If the Control Room indication cannot be restored within 30 days, the reactor shall be in hot shutdown within six hours and in cold shutdown within the subsequent 24 hours.

TABLE 3.2.8

EMERGENCY BUS UNDERVOLTAGE INSTRUMENTATION

Minimum Number of Operable Instruments	Parameter	Trip Setting	Required Action
2 per bus	Degraded Bus Voltage - Voltage (27/3Z, 27/3W, 27/4Z, 27/4W)	3,700 volts ± 40 volts	Note 1
2 per bus	Degraded Bus Voltage - Time Delay (62/3W, 62/3Z, 62/4W, 62/4Z)	10 seconds ± 1 second	Note 2

TABLE 3.2.7 NOTES

1. If the minimum number of operable instrument channels are not available, the inoperable channel shall be tripped using test jacks or other permanently installed circuits within one hour.

2. If the minimum number of operable instrument channels are not available, reactor power operation is permissible for only 7 successive days unless the system is sooner made operable.

TABLE 3.2.9

REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

Minimum Number of Operable Instrument Channels per Trip System	Trip Function	Trip Level Setting	Required Action When Minimum Conditions for Operation Are Not Satisfied
2 (Note 1)	Low-Low Reactor Vessel Water Level	≥82.5" Above Top of Enriched Fuel	Note 4
2 (Note 2)	Low Condensate Storage Tank Water Level	≥3%	Note 4
2 (Note 3)	High Reactor Vessel Water Level	≤177" Above Top of Enriched Fuel	Note 4
1	Bus Power Monitor	--	Note 4
1	Trip System Logic	--	Note 4

TABLE 3.2.9 NOTES

1. One trip system with initiating instrumentation arranged in a one-out-of-two taken twice logic.

2. One trip system with initiating instrumentation arranged in a one-out-of-two logic.

3. One trip system arranged in a two-out-of-two logic.

4. If the minimum number of operable channels are not available, the system is considered inoperable and the requirements of Specification 3.5 apply.

TABLE 4.2.1

MINIMUM TEST AND CALIBRATION FREQUENCIES

EMERGENCY CORE COOLING ACTUATION INSTRUMENTATION

Core Spray System

Trip Function	Functional Test(8)	Calibration(8)	Instrument Check
High Drywell Pressure	(Note 1)	Once/Operating Cycle	Once Each Day
Low-Low Reactor Vessel Water Level	(Note 1)	Once/Operating Cycle	Once Each Day
Low Reactor Pressure (PT-2-3-56C/D(S1))	(Note 1)	Once/Operating Cycle	--
Low Reactor Pressure (PT-2-3-56A/B(S1) & 52C/D(M))	(Note 1)	Once/Operating Cycle	--
Pump 14-1A, Discharge Pressure	(Note 1)	Every Three Months	--
Auxiliary Power Monitor	(Note 1)	Every Refueling	Once Each Day
Pump Bus Power Monitor	(Note 1)	None	Once Each Day
High Sparger Pressure	(Note 1)	Every Three Months	--
Trip System Logic	Once/Operating Cycle	Once/Operating Cycle (Note 3)	--

TABLE 4.2.1
(Cont'd)

MINIMUM TEST AND CALIBRATION FREQUENCIES

EMERGENCY CORE COOLING ACTUATION INSTRUMENTATION

Low Pressure Coolant Injection System			
Trip Function	Functional Test(8)	Calibration(8)	Instrument Check
Low Reactor Pressure (PT-2-3-56C/D(S1))	(Note 1)	Once/Operating Cycle	--
High Drywell Pressure (PT-10-101A-D(M))	(Note 1)	Once/Operating Cycle	Once Each Day
Low-Low Reactor Vessel Water Level	(Note 1)	Once/Operating Cycle	Once Each Day
Reactor Vessel Shroud Level	(Note 1)	Once/Operating Cycle	--
Low Reactor Pressure (PT-2-128A/B)	(Note 1)	Every Three Months	--
RHR Pump Discharge Pressure	(Note 1)	Every Three Months	--
High Drywell Pressure (PT-10-101A-D(S1))	(Note 1)	Once/Operating Cycle	--
Low Reactor Pressure (PT-2-3-56A/B)(S1) & 52C/D(M))	(Note 1)	Once/Operating Cycle	--
Auxiliary Power Monitor	(Note 1)	Every Refueling Outage	Once Each Day
Pump Bus Power Monitor	(Note 1)	None	Once Each Day
Trip System Logic	Once/Operating Cycle	Once/Operating Cycle (Note 3)	--

TABLE 4.2.1
(Cont'd)

MINIMUM TEST AND CALIBRATION FREQUENCIES

EMERGENCY CORE COOLING ACTUATION INSTRUMENTATION

High Pressure Coolant Injection System			
Trip Function	Functional Test(8)	Calibration(8)	Instrument Check
Low-Low Reactor Vessel Water Level	(Note 1)	Once/operating cycle	Once each day
Low Condensate Storage Tank Water Level	(Note 1)	Every three months	--
High Drywell Pressure	(Note 1)	Once/operating cycle	Once each day
Bus Power Monitor	(Note 1)	None	Once each day
Trip System Logic	Once/operating cycle	Once/Operating cycle (Note 3)	--
High Reactor Vessel Water Level	(Note 1)	Once/operating cycle	--

TABLE 4.2.1
(Cont'd)

MINIMUM TEST AND CALIBRATION FREQUENCIES

EMERGENCY CORE COOLING ACTUATION INSTRUMENTATION

Automatic Depressurization System

Trip Function	Functional Test(8)	Calibration(8)	Instrument Check
Low-Low Reactor Vessel Water Level	(Note 1)	Once/Operating Cycle	Once Each Day
High Drywell Pressure	(Note 1)	Once/Operating Cycle	Once Each Day
Bus Power Monitor	(Note 1)	None	Once Each Day
Trip System Logic (Except Solenoids of Valves)	Once/Operating Cycle (Notes 2 and 11)	Once/Operating Cycle (Note 3)	--

TABLE 4.2.1
(Cont'd)

MINIMUM TEST AND CALIBRATION FREQUENCIES

EMERGENCY CORE COOLING ACTUATION INSTRUMENTATION

Recirculation Pump Trip Actuation System			
Trip Function	Functional Test(8)	Calibration(8)	Instrument Check
Low-Low Reactor Vessel Water Level (4)	(Note 1)	Once/Operating Cycle	Once Each Day
High Reactor Pressure (4)	(Note 1)	Once/Operating Cycle	Once Each Day
Trip System Logic	Once/Operating Cycle	Once/Operating Cycle	--

TABLE 4.2.2

MINIMUM TEST AND CALIBRATION FREQUENCIES

PRIMARY CONTAINMENT ISOLATION INSTRUMENTATION

Trip Function	Functional Test(8)	Calibration(8)	Instrument Check
Low-Low Reactor Vessel Water Level	(Note 1)	Once/Operating Cycle	Once Each Day
High Steam Line Area Temperature	(Note 1)	Each Refueling Outage	--
High Steam Line Flow	(Note 1)	Once/Operating Cycle	Once Each Day
Low Main Steam Line Pressure	(Note 1)	Every Three Months	--
Low Reactor Vessel Water Level	(Note 1)	Once/Operating Cycle	--
High Main Steam Line Radiation	(Notes 1 and 7)	Each Refueling Outage	Once Each Day
High Drywell Pressure	(Note 1)	Once/Operating Cycle	Once Each Day
Condenser Low Vacuum	(Note 1)	Every Three Months	--
Trip System Logic	Once/Operating Cycle (Note 2)	Once/Operating Cycle (Note 3)	--

TABLE 4.2.2
(Cont'd)

MINIMUM TEST AND CALIBRATION FREQUENCIES

HIGH PRESSURE COOLANT INJECTION SYSTEM ISOLATION INSTRUMENTATION

Trip Function	Functional Test(8)	Calibration(8)	Instrument Check
High Steam Line Space Temperature	(Note 1)	Each Refueling Outage	--
High Steam Line D/P (Steam Line Break)	(Note 1)	Every Three Months	--
Low HPCI Steam Supply Pressure	(Note 1)	Every Three Months	--
Main Steam Line Tunnel Temperature	(Note 1)	Each Refueling Outage	--
Bus Power Monitor	(Note 1)	None	Once Each Day
Trip System Logic	Once/Operating Cycle	Once/Operating Cycle (Note 3)	--

TABLE 4.2.2
(Cont'd)

MINIMUM TEST AND CALIBRATION FREQUENCIES

REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION INSTRUMENTATION

Trip Function	Functional Test(8)	Calibration(8)	Instrument Check
Main Steam Line Tunnel Temperature	(Note 1)	Each Refueling Outage	--
High Steam Line Space Temperature	(Note 1)	Each refueling outage	--
High Steam Line d/p including time delay relays (Steam Line Break)	(Note 1)	Every three months	--
Low RCIC Steam Supply Pressure	(Note 1)	Every three months	--
Bus Power Monitor	(Note 1)	None	Once each day
Trip System Logic	Once/operating cycle	Once/operating cycle (Note 3)	--

TABLE 4.2.3

MINIMUM TEST AND CALIBRATION FREQUENCIES

REACTOR BUILDING VENTILATION AND STANDBY GAS TREATMENT SYSTEM ISOLATION

Trip Function	Functional Test(8)	Calibration(8)	Instrument Check
Low Reactor Vessel Water Level	(Note 1)	Once/Operating Cycle	--
High Drywell Pressure	(Note 1)	Once/Operating Cycle	--
Reactor Building Vent Exhaust Radiation	Monthly	Every Three Months	Once Each Day
Refueling Floor Zone Radiation	Monthly	Every Three Months	Once Each Day During Refueling
Reactor Building Vent Trip System Logic	Once/Operating Cycle	Once/Operating Cycle (Note 3)	--
Standby Gas Treatment Trip System Logic	Once/Operating Cycle	Once/Operating Cycle (Note 3)	--
Logic Bus Power Monitor	(Note 1)	None	Once Each Day

TABLE 4.2.4

MINIMUM TEST AND CALIBRATION FREQUENCIES

OFF-GAS SYSTEM ISOLATION INSTRUMENTATION

Trip Function	Functional Test(8)	Calibration(8)	Instrument Ch
Augmented Off-Gas Trip System Logic (AOG)	Once/Operating Cycle (Note 2)	Once/Operating Cycle (Note 3)	--

TABLE 4.2.5

MINIMUM TEST AND CALIBRATION FREQUENCIES

CONTROL ROD BLOCK INSTRUMENTATION

Trip Function	Functional Test	Calibration
Startup Range Monitor		
a. Upscale	Notes 4 and 6	Note 6
b. Detector Not Fully Inserted	Note 6	N/A
Intermediate Range Monitor		
a. Upscale	Notes 4 and 6	Note 6
b. Downscale	Notes 4 and 6	Note 6
c. Detector Not Fully Inserted	Note 6	N/A
Average Power Range Monitor		
a. Upscale (Flow Bias)	Notes 1 and 4	Every Three Months
b. Downscale	Notes 1 and 4	Every Three Months
Rod Block Monitor		
a. Upscale (Flow Bias)	Notes 1 and 4	Every Three Months
b. Downscale	Notes 1 and 4	Every Three Months
Trip System Logic	Once/Operating Cycle	Once/Operating Cycle (Note 3)
High Water Level in Scram Discharge Volume	Every Three Months	Refueling Outage

TABLE 4.2.6

CALIBRATION REQUIREMENTS

POST-ACCIDENT INSTRUMENTATION

Parameter	Calibration	Instrument Check
Drywell Atmosphere Temperature	Every 6 Months	Once Each Day
Containment Pressure	Once/Operating Cycle	Once Each Day
Torus Pressure	Once/Operating Cycle	Once Each Day
Torus Water Level	Once/Operating Cycle	Once Each Day
Torus Water Temperature	Every 6 Months	Once Each Day
Reactor Pressure	Once/Operating Cycle	Once Each Day
Reactor Vessel Water Level	Once/Operating Cycle	Once Each Day
Control Rod Position	(Note 5)	Once Each Day
Neutron Monitor	Same as Reactor Protection Systems	Once Each Day
Torus Air Temperature	Every 6 Months	Once Each Day
Safety/Relief Valve Position	Every Refueling Outage (Note 9) (a Functional Test to be performed quarterly)	Once Each Day
Safety Valve Position	Every Refueling Outage (Note 9) (a Functional Test to be performed quarterly)	Once Each Day

TABLE 4.2.6
(Cont'd)

CALIBRATION REQUIREMENTS

POST-ACCIDENT INSTRUMENTATION

Parameter	Calibration	Instrument Check
Containment Hydrogen/Oxygen Monitor	Once/Operating Cycle	Once each day
Containment High-Range Radiation Monitor	Once/Operating Cycle	Once each day
Stack Noble Gas Effluent	Every Operating Cycle (a Functional Test to be performed quarterly)	Once each day

TABLE 4.2.8

EMERGENCY BUS UNDERVOLTAGE INSTRUMENTATION

Trip System	Functional Test	Calibration (8)
Degraded Bus Voltage	See Note 10	Once/Operating Cycle

TABLE 4.2.9

MINIMUM TEST AND CALIBRATION FREQUENCIES

REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

Trip Function	Functional Test(8)	Calibration(8)	Instrument Check
Low-Low Reactor Vessel Water Level	(Note 1)	Once/Operating Cycle	Once each day
Low Condensate Storage Tank Water Level	(Note 1)	Once/Operating Cycle	--
High Reactor Vessel Water Level	(Note 1)	Once/Operating Cycle	--
Bus Power Monitor	(Note 1)	None	Once each day
Trip System Logic	Once/Operating Cycle	Once/Operating Cycle (Note 3)	--

TABLE 4.2 NOTES

1. Initially once per month; thereafter, a longer interval as determined by test results on this type of instrumentation.

2. During each refueling outage, simulated automatic actuation which opens all pilot valves shall be performed such that each trip system logic can be verified independent of its redundant counterpart.

3. Trip system logic calibration shall include only time delay relays and timers necessary for proper functioning of the trip system.

4. This instrumentation is expected from functional test definition. The functional test will consist of injecting a simulated electrical signal into the measurement channel.

5. Check control rod position indication while performing the surveillance requirement of Section 3.3.

6. Functional tests, calibrations, and instrument checks are not required when these instruments are not required to be operable or are tripped. Functional tests shall be performed before each startup with a required frequency not to exceed once per week. Calibration shall be performed prior to or during each startup or controlled shutdown with a required frequency not to exceed once per week. Instrument checks shall be performed at least once per day during those periods when instruments are required to be operable.

7. This instrumentation is excepted from the functional test definitions and shall be calibrated using simulated electrical signals once every three months.

8. Functional tests and calibrations are not required when systems are not required to be operable.

9. The thermocouples associated with safety/relief valves and safety valve position, that may be used for back-up position indication, shall be verified to be operable every operating cycle.

10. Separate functional tests are not required for this instrumentation. The calibration and integrated ECCS tests which are performed once per operating cycle will adequately demonstrate proper equipment operation.

11. Trip system logic functional tests will include verification of operation of all automatic initiation inhibit switches by monitoring relay contact movement. Verification that the manual inhibit switches prevent opening all relief valves will be accomplished in conjunction with Section 4.5.F.1.

BASES:

3.2 PROTECTIVE INSTRUMENTATION

In addition to reactor protection instrumentation which initiates a reactor scram, station protective instrumentation has been provided which initiates action to mitigate the consequences of accidents which are beyond the reactor operator's ability to control, or terminate a single operator error before it results in serious consequences. This set of Specifications provides the limiting conditions of operation for the primary system isolation function and initiation of the core standby cooling and standby gas treatment systems. The objectives of the Specifications are (i) to assure the effectiveness of any component of such systems even during periods when portions of such systems are out of service for maintenance, testing, or calibration, and (ii) to prescribe the trip settings required to assure adequate performance. This set of Specifications also provides the limiting conditions of operation for the control rod block system and surveillance instrumentation.

Isolation valves (Note 1) are installed in those lines that penetrate the primary containment and must be isolated during a loss-of-coolant accident so that the radiation dose limits are not exceeded during an accident condition. Actuation of these valves is initiated by protective instrumentation shown in Table 3.2.2 which senses the conditions for which isolation is required. Such instrumentation must be available whenever primary containment integrity is required. The objective is to isolate the primary containment so that the limits of 10 CFR 100 are not exceeded during an accident. The objective of the low turbine condenser vacuum trip is to minimize the radioactive effluent releases to as low as practical in case of a main condenser failure. Subsequent releases would continue until operator action was taken to isolate the main condenser unless the main steam line isolation valves were closed automatically on low condenser vacuum. The manual bypass is required to permit initial startup of the reactor during low power operation.

The instrumentation which initiates primary system isolation is connected in a dual channel arrangement. Thus, the discussion given in the bases for Specification 3.1 is applicable here.

The low reactor water level instrumentation is set to trip when reactor water level is 127" above the top of the enriched fuel. This trip initiates closure of Group 2 and 3 primary containment isolation valves. For a trip setting of 127" above the top of the enriched fuel, the valves will be closed before perforation of the clad occurs even for the maximum break and, therefore, the setting is adequate.

The top of the enriched fuel (351.5" from vessel bottom) is designated as a common reference level for all reactor water level instrumentation. The intent is to minimize the potential for operator confusion which may result from different scale references.

The low-low reactor water level instrumentation is set to trip when reactor water level is 82.5" H_2O indicated on the reactor water level instrumentation above the top of the enriched fuel. This trip initiates closure of the Group 1 primary containment isolation valves and also activates the ECCS and RCIC System and starts the standby diesel generator system. This trip setting level was chosen to be low enough to prevent spurious operation, but high enough to initiate ECCS operation and primary system isolation so that no melting of the fuel cladding will occur, and so that post-accident cooling can be accomplished and the limits of 10CFR100 will not be violated. For the complete circumferential break of 28-inch recirculation line and with the trip

Note 1 - Isolation valves are grouped as listed in Table 3.7.1.

BASES: 3.2 (Cont'd)

setting given above, ECCS initiation and primary system isolation are initiated in time to meet the above criteria. The instrumentation also covers the full range of spectrum breaks and meets the above criteria.

The high drywell pressure instrumentation is a backup to the water level instrumentation, and in addition to initiating ECCS, it causes isolation of Group 2, 3, and 4 isolation valves. For the complete circumferential break discussed above, this instrumentation will initiate ECCS operation at about the same time as the low-low water level instrumentation, thus, the results given above are applicable here also. Group 2 isolation valves include the drywell vent, purge, and sump isolation valves. High drywell pressure activates only these valves because high drywell pressure could occur as the result of nonsafety-related causes such as not purging the drywell air during startup. Total system isolation is not desirable for these conditions and only the valves in Group 2 are required to close. The water level instrumentation initiates protection for the full spectrum of loss-of-coolant accidents and causes a trip of all primary system isolation valves.

Venturis are provided in the main steam lines as a means of measuring steam flow and also limiting the loss of mass inventory from the vessel during a steam line break accident. In addition to monitoring steam flow, instrumentation is provided which causes a trip of Group 1 isolation valves. The primary function of the instrumentation is to detect a break in the main steam line, thus only Group 1 valves are closed. For the worst case accident, main steam line break outside the drywell, this trip setting of 140 percent of rated steam flow in conjunction with the flow limiters and main steam line valve closure limit the mass inventory loss such that fuel is not uncovered, cladding temperatures remain less than 1295°F and release of radioactivity to the environs is well below 10CFR100.

Temperature monitoring instrumentation is provided in the main steam line tunnel to detect leaks in this area. Trips are provided on this instrumentation and when exceeded cause closure of Group 1 isolation valves. Its setting of ambient plus 95°F is low enough to detect leaks of the order of 5 to 10 gpm; thus, it is capable of covering the entire spectrum of breaks. For large breaks, it is a backup to high steam flow instrumentation discussed above, and for small breaks, with the resultant small release of radioactivity, gives isolation before the limits of 10CFR100 are exceeded.

High radiation monitors in the main steam line tunnel have been provided to detect gross fuel failure resulting from a control rod drop accident. This instrumentation causes closure of Group 1 valves, the only valves required to close for this accident. With the established setting of 3 times normal background and main steam line isolation valve closure, fission product release is limited so that 10CFR100 limits are not exceeded for the control rod drop accident, and 10CFR20 limits are not exceeded for gross fuel failure during reactor operations. With an alarm setting of 1.5 times normal background, the operator is alerted to possible gross fuel failure or abnormal fission product releases from failed fuel due to transient reactor operation.

Pressure instrumentation is provided which trips when main steam line pressure drops below 800 psig. A trip of this instrumentation results in closure of Group 1 isolation valves. In the refuel, shutdown, and startup modes, this trip function is provided when main steam line flow exceeds 40% of rated capacity. This function is provided primarily to provide protection against a pressure regulator malfunction which would cause the control and/or bypass valves to open, resulting in a rapid depressurization and cooldown of the reactor vessel. The 800 psig trip

BASES: 3.2 (Cont'd)

setpoint limits the depressurization such that no excessive vessel thermal stress occurs as a result of a pressure regulator malfunction. This setpoint was selected far enough below normal main steam line pressures to avoid spurious primary containment isolations.

Low condenser vacuum has been added as a trip of the Group 1 isolation valves to prevent release of radioactive gases from the primary coolant through condenser. The setpoint of 12 inches of mercury absolute was selected to provide sufficient margin to assure retention capability in the condenser when gas flow is stopped and sufficient margin below normal operating values.

The HPCI and/or RCIC high flow, steam supply pressure, and temperature instrumentation is provided to detect a break in the HPCI and/or RCIC piping. Tripping of this instrumentation results in actuation of HPCI and/or RCIC isolation valves, i.e., Group 6 valves. A time delay has been incorporated into the RCIC steam flow trip logic to prevent the system from inadvertently isolating due to pressure spikes which may occur on startup. The trip settings are such that core uncovering is prevented and fission product release is within limits.

The instrumentation which initiates ECCS action is arranged in a dual channel system. Permanently installed circuits and equipment may be used to trip instrument channels. In the nonfail safe systems which require energizing the circuitry, tripping an instrument channel may take the form of providing the required relay function by use of permanently installed circuits. This is accomplished in some cases by closing logic circuits with the aid of the permanently installed test jacks or other circuitry which would be installed for this purpose.

The control rod block functions are provided to prevent excessive control rod withdrawal so that MCPR does not decrease below the fuel cladding integrity safety limit. The trip logic for this function is 1 out of n; e.g., any trip on one of the six APRMs, six IRMs or four SRMs will result in a rod block. The minimum instrument channel requirements for the IRM may be reduced by one for a short period of time to allow for maintenance, testing or calibration. The RBM is an operational guide and aid only and is not needed for rod withdrawal.

For single recirculation loop operation, the RBM trip setting is reduced in accordance with the analysis presented in NEDO-30060, February 1983. This adjustment accounts for the difference between the single loop and two loop drive flow at the same core flow, and ensures that the margin of safety is not reduced during single loop operation.

The APRM rod block trip is flow referenced and prevents a significant reduction in MCPR especially during operation at reduced flow. The APRM provides gross core protection; i.e., limits the gross core power increase from withdrawal of control rods in the normal withdrawal sequence. The trips are set so that MCPR is maintained greater than the fuel cladding integrity safety limit. For single recirculation loop operation, the APRM rod block trip setting is reduced in accordance with the analysis presented in NEDE-30060, February 1983. This adjustment accounts for the difference between the single loop and two-loop drive flow at the same core flow, and ensures that the margin of safety is not reduced during single loop operation.

The IRM rod block function provides local as well as gross core protection. The scaling arrangement is such that trip setting is less than a factor of 10 above the indicated level. Analysis of the worst-case accident results in rod block action before MCPR approaches the fuel cladding integrity safety limit.

BASES: 3.2 (Cont'd)

A downscale indication on an APRM or IRM is an indication the instrument has failed or the instrument is not sensitive enough. In either case, the instrument will not respond to changes in control rod motion and thus control rod motion is prevented.

To prevent excessive clad temperatures for the small pipe break, the HPCI or Automatic Depressurization System must function since, for these breaks, reactor pressure does not decrease rapidly enough to allow either core spray or LPCI to operate in time. For a break or other event occurring outside the drywell, the Automatic Depressurization System is initiated on low-low reactor water level only after a time delay. The arrangement of the tripping contacts is such as to provide this function when necessary and minimize spurious operation. The trip settings given in the Specification are adequate to ensure the above criteria are met. The Specification preserves the effectiveness of the system during periods of maintenance, testing, or calibration, and also minimizes the risk of inadvertent operation; i.e., only one instrument channel out of service.

The ADS is provided with inhibit switches to manually prevent automatic initiation during events where actuation would be undesirable, such as certain ATWS events. The system is also provided with an Appendix R inhibit switch to prevent inadvertent actuation of ADS during a fire which requires evacuation of the Control Room.

Four radiation monitors are provided which initiate isolation of the reactor building and operation of the standby gas treatment system. The monitors are located in the reactor building ventilation duct and on the refueling floor. Any one upscale trip or two downscale trips of either set of monitors will cause the desired action. Trip settings for the monitors on the refueling floor are based upon initiating normal ventilation isolation and standby gas treatment system operation so that none of the activity released during the refueling accident leave the Reactor Building via the normal ventilation stack but that all activity is processed by the standby gas treatment system. Trip settings for the monitors in the ventilation duct are based upon initiation of the normal ventilation isolation and standby gas treatment system operation at a radiation level equivalent to the maximum site boundary dose rate of 500 mrem/year as given in Specification 3.8.E.1.a. The monitoring system in the plant stack represents a backup to this system to limit gross radioactivity releases to the environs.

The purpose of isolating the mechanical vacuum pump line is to limit release of radioactivity from the main condenser. During an accident, fission products would be transported from the reactor through the main steam line to the main condenser. The fission product radioactivity would be sensed by the main steam line radiation monitors which initiate isolation.

Post-accident instrumentation parameters for Containment Pressure, Torus Water Level, Containment Hydrogen/Oxygen Monitor, and Containment High-Range Radiation Monitor, are redundant, environmentally and seismically qualified instruments provided to enhance the operators' ability to follow the course of an event. The purpose of each of these instruments is to provide detection and measurement capability during and following an accident as required by NUREG-0737 by ensuring continuous on-scale indication of the following: containment pressure in the 0 to 275 psia range; torus water level in the 0 to 25 foot range (i.e., the bottom to 5 feet above the normal water level of the torus pool); containment hydrogen/oxygen concentrations (0 to 30% hydrogen and 0 to 25% oxygen); and containment radiation in the 1 R/hr to 10^7 R/hr gamma.

BASES: 3.2 (Cont'd)

The Degraded Grid Protective System has been installed to assure that safety-related electrical equipment will not be subjected to sustained degraded voltage. This system incorporates voltage relays on 4160 Volt Emergency Buses 3 and 4 which are set to actuate at the minimum voltage required to prevent damage of safety-related equipment.

If Degraded Grid conditions exist for 10 seconds, either relay will actuate an alarm to alert operators of this condition. Based upon an assessment of these conditions the operator may choose to manually disconnect the off-site power. In addition, if an ESF signal is initiated in conjunction with low voltage below the relay setpoint for 10 seconds, the off-site power will be automatically disconnected.

The Reactor Core Isolation Cooling (RCIC) System provides makeup water to the reactor vessel during shutdown and isolation to supplement or replace the normal makeup sources without the use of the Emergency Core Cooling Systems. The RCIC System is initiated automatically upon receipt of a reactor vessel low-low water level signal. Reactor vessel high water level signal results in shutdown of the RCIC System. However, the system will restart on a subsequent reactor vessel low-low water level signal. The RCIC System is normally lined up to take suction from the condensate storage tank. Suction will automatically switch over from the condensate storage tank to the suppression pool on low condensate storage tank level.

BASES:

4.2 PROTECTIVE INSTRUMENTATION

The Protective Instrumentation Systems covered by this Specification are listed in Table 4.2. Most of these protective systems are composed of two or more independent and redundant subsystems which are combined in a dual-channel arrangement. Each of these subsystems contains an arrangement of electrical relays which operate to initiate the required system protective action.

The relays in a subsystem are actuated by a number of means, including manually-operated switches, process-operated switches (sensors), bistable devices operated by analog sensor signals, timers, limit switches, and other relays. In most cases, final subsystem relay actuation is obtained by satisfying the logic conditions established by a number of these relay contacts in a logic array. When a subsystem is actuated, the final subsystem relay(s) can operate protective equipment, such as valves and pumps, and can perform other protective actions, such as tripping the main turbine generator unit.

With the dual-channel arrangement of these subsystems, the single failure of a ready circuit can be tolerated because the redundant subsystem or system (in the case of high pressure coolant injection) will then initiate the necessary protective action. If a failure in one of these circuits occurs in such a way that an action is taken, the operator is immediately alerted to the failure. If the failure occurs and causes no action, it could then remain undetected, causing a loss of the redundancy in the dual-channel arrangement. Losses in redundancy of this nature are found by periodically testing the relay circuits and contacts in the subsystems to assure that they are operating properly.

It has been the practice in boiling water reactor plants to functionally test protective instrumentation sensors and sensor relays on-line on a monthly frequency. Since logic circuit tests result in the actuation of plant equipment, testing of this nature was done while the plant was shut down for refueling. In this way, the testing of equipment would not jeopardize plant operation.

This Specification is a periodic testing program which is based upon the overall testing of protective instrumentation systems, including logic circuits as well as sensor circuits. Table 4.2 outlines the test, calibration, and logic system functional test schedule for the protective instrumentation systems. The testing of a subsystem includes a functional test of each relay wherever practicable. The testing of each relay includes all circuitry necessary to make the relay operate, and also the proper functioning of the relay contacts. Testing of the automatic initiation inhibit switches verifies the proper operability of the switches and relay contacts. Functional testing of the inaccessible temperature switches associated with the isolation systems is accomplished remotely by application of a heat source to individual switches.

All subsystems are functionally tested, calibrated, and operated in their entirety.

3.3 LIMITING CONDITIONS FOR OPERATION

3.3 CONTROL ROD SYSTEM

Applicability:

Applies to the operational status of the control rod system.

Objective:

To assure the ability of the control rod system to control reactivity.

Specification:

A. Reactivity Limitations

1. Reactivity Margin - Core Loading

The core loading shall be limited to that which can be made subcritical in the most reactive condition during the operation cycle with the highest worth, operable control rod in its fully withdrawn position and all other operable rods inserted.

2. Reactivity Margin - Inoperable Control Rods

Control rod driven which cannot be moved with control rod drive pressure shall be considered inoperable. If a partially or fully withdrawn control rod drive cannot be moved with drive or scram pressure, the reactor shall be brought to a shutdown condition within 48 hours unless investigation demonstrates that the cause of the failure is not due to a failed control rod drive mechanism collet housing. The control rod directional control valves for inoperable control rods shall be

4.3 SURVEILLANCE REQUIREMENTS

4.3 CONTROL ROD SYSTEM

Applicability:

Applies to the surveillance requirements of the control rod system.

Objective:

To verify the ability of the control rod system to control reactivity.

Specification:

A. Reactivity Limitations

1. Reactivity Margin - Core Loading

Control rods shall be withdrawn following a refueling outage when core alterations were performed to demonstrate a shutdown margin of 0.25 per cent Δk at any time in the subsequent fuel cycle with the highest worth operable control rod fully withdrawn and all other operable rods inserted.

2. Reactivity Margin - Inoperable Control Rods

Each partially or fully withdrawn operable control rod shall be exercised one notch at least once each week. This test shall be performed at least once per 24 hours in the event power operation is continuing with two or more inoperable control rods or in the event power operation is continuing with one fully or partially withdrawn rod which cannot be moved and for which control rod drive mechanism damage has not been ruled out. The surveillance need not be completed within 24 hours if the number

3.3 LIMITING CONDITIONS FOR OPERATION

disarmed electrically except for control rods which are inoperable because of scram times greater than those specified in Specification 3.3.C. In no case shall the number of inoperable rods which are not fully inserted be greater than six during power operation.

B. Control Rods

1. Each control rod shall be either coupled to its drive or placed in the inserted position and its directional valves disarmed electrically. When removing up to one control rod drive per quadrant for inspection and the reactor is in the refueling mode, this requirement does not apply.

4.3 SURVEILLANCE REQUIREMENTS

of inoperable rods has been reduced to less than two and if it has been demonstrated that control rod drive mechanism collet housing failure is not the cause of an immovable control rod.

B. Control Rods

1. The coupling integrity shall be verified:

   (a) When a rod is withdrawn the first time subsequent to each refueling outage or after maintenance, observe discernable response of the nuclear instrumentation; however, for initial rods when response is not discernable, subsequent exercising of these rods after the reactor is critical shall be performed to verify instrumentation response; and

   (b) When a rod is fully withdrawn, observe that the rod does not go to the over-travel position. Prior to startup following a refueling outage, each rod shall be fully withdrawn continuously to observe that the rate of withdrawal is proper and that the rod does not go to the over-travel position. Following uncoupling, each control rod drive and blade shall be tested to verify

3.3 LIMITING CONDITIONS FOR OPERATION

4.3 SURVEILLANCE REQUIREMENTS

positive coupling and the results of each test shall be recorded. This test shall consist of checking the operability of the over-travel indicator circuit prior to coupling by withdrawing the drive and observing the over-travel light. The drive and blade shall then be immediately coupled and fully withdrawn. The position and over-travel lights shall be observed.

2. The Control Rod Drive Housing Support System shall be in place when the Reactor Coolant System is pressurized above atmospheric pressure with fuel in the reactor vessel unless all operable control rods are fully inserted.

2. The Control Rod Drive Housing Support System shall be inspected after reassembly and the results of the inspection recorded.

3. While the reactor is below 20% power, the Rod Worth Minimizer (RWM) shall be operating while moving control rods except that:

   (a) If after withdrawal of at least 12 control rods during a startup, the RWM fails, the startup may continue provided a second licensed operator verifies that the operator at the reactor console is following the control rod program; or

   (b) If all rods, except those that cannot be moved with control rod drive

3. Prior to control rod withdrawal for startup the Rod Worth Minimizer (RWM) shall be verified as operable by performing the following:

   (a) The Reactor Engineer shall verify that the control rod withdrawal sequence for the Rod Worth Minimizer computer is correct.

   (b) The Rod Worth Minimizer diagnostic test shall be performed.

3.3 LIMITING CONDITIONS FOR OPERATION

pressure, are fully inserted, no more than two rods may be moved.

4. Control rod patterns and the sequence of withdrawal or insertion shall be established such that the rod drop accident limit of 280 cal/g is not exceeded.

5. Control rods shall not be withdrawn for startup or refueling unless at least two source range channels have an observed count rate greater than or equal to three counts per second.

6. During operation with limiting control rod patterns either:

   (a) Both RBM channels shall be operable; or

   (b) Control rod withdrawal shall be blocked; or

4.3 SURVEILLANCE REQUIREMENTS

(c) Out-of-sequence control rods in each distinct RWM group shall be selected and the annunciator of the selection errors verified.

(d) An out-of-sequence control rod shall be withdrawn no more than three notches and the rod block function verified.

4. The control rod pattern and sequence of withdrawal or insertion shall be verified to comply with Specification 3.3.B.4.

5. Prior to control rod withdrawal for startup or during refueling, verification shall be made that at least two source range channels have an observed count rate of at least three counts per second.

6. When a limiting control rod pattern exists, an instrument functional test of the RBM shall be performed prior to withdrawal of the designated rod(s) and daily thereafter.

3.3 LIMITING CONDITIONS FOR OPERATION

(c) The operating power level shall be limited so that the MCPR will remain above the fuel cladding integrity safety limit assuming a single error that results in complete withdrawal of any single operable control rod.

C. Scram Insertion Times

1.1 The average scram time, based on the de-energization of the scram pilot valve solenoids of all operable control rods in the reactor power operation condition shall be no greater than:

Drop-Out of Position	% Inserted From Fully Withdrawn	Avg. Scram Insertion Time (sec)
46	4.51	0.358
36	25.34	0.912
26	46.18	1.468
06	87.84	2.686

The average of the scram insertion times for the three fastest control rods of all groups of four control rods in a two by two array shall be no greater than:

Drop-Out of Position	% Inserted From Fully Withdrawn	Avg. Scram Insertion Time (sec)
46	4.51	0.379
36	25.34	0.967
26	46.18	1.556
06	87.84	2.848

4.3 SURVEILLANCE REQUIREMENTS

7. The scram discharge volume drain and vent valves shall be verified open at least once per month. These valves may be closed intermittently for testing under administrative control.

C. Scram Insertion Times

1. After refueling outage and prior to operation above 30% power with reactor pressure above 800 psig all control rods shall be subject to scram-time measurements from the fully withdrawn position. The scram times for single rod scram testing shall be measured without reliance on the control rod drive pumps.

2. During or following a controlled shutdown of the reactor, but not more frequently than 16 weeks nor less frequently than 32 weeks intervals, 50% control rod drives in each quadrant of the reactor core shall be measured for scram times specified in Specification 3.3.C. All control rod drives shall have experienced scram-time measurements each year. Whenever 50% of the control rod drives scram times have been measured, an evaluation shall be made to provide reasonable assurance that proper control rod drives performance is being maintained. The results of measurements performed on the control rod drives shall be submitted in the start up test report.

3.3 LIMITING CONDITIONS FOR OPERATION

4.3 SURVEILLANCE REQUIREMENTS

1.2 If Specification 3.3.C.1.1 cannot be met, the average scram time, based on the de-energization of the scram pilot valve solenoids of all operable control rods in the reactor power operation condition shall be no greater than:

Drop-Out of Position	% Inserted From Fully Withdrawn	Avg. Scram Insertion Time (sec)
46	4.51	.358
36	25.34	1.096
26	46.18	1.860
06	87.84	3.419

The average of the scram insertion times for the three fastest control rods of all groups of four control rods in a two by two array shall be no greater than:

Drop-Out of Position	% Inserted From Fully Withdrawn	Avg. Scram Insertion Time (sec)
46	4.51	.379
36	25.34	1.164
26	46.18	1.971
06	87.84	3.624

2. The maximum scram insertion time for 90% insertion of any operable control rod shall not exceed 7.00 seconds.

3.3 LIMITING CONDITIONS FOR OPERATION

3. If Specification 3.3.C.1.2 cannot be met, the reactor shall not be made supercritical; if operating, the reactor shall be shut down immediately upon determination that average scram time is deficient.

4. If Specification 3.3.C.2 cannot be met, the deficient control rod shall be considered inoperable, fully inserted into the core, and electrically disarmed.

D. Control Rod Accumulators

At all reactor operating pressures, a rod accumulator may be inoperable provided that no other control rod in the nine-rod square array around this rod has a:

1. Inoperable accumulator.

2. Directional control valve electrically disarmed while in a non-fully inserted position.

3. Scram insertion greater than maximum permissible insertion time.

If a control rod with an inoperable accumulator is inserted "full-in" and its directional control valves are electrically disarmed, it shall not be considered to have an inoperable accumulator.

4.3 SURVEILLANCE REQUIREMENTS

D. Control Rod Accumulators

Once a shift check the status of the pressure and level alarms for each accumulator.

3.3 LIMITING CONDITIONS FOR OPERATION

E. Reactivity Anomalies

The reactivity equivalent of the difference between the actual critical rod configuration and the expected configuration during power operation shall not exceed 1% Δk. If this limit is exceeded, the reactor will be shut down until the cause has been determined and corrective actions have been taken if such actions are appropriate.

F. If Specification 3.3A through D above are not met, an orderly shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours.

4.3 SURVEILLANCE REQUIREMENTS

E. Reactivity Anomalies

During the startup test program and startups following refueling outages, the critical rod configurations will be compared to the expected configurations at selected operating conditions. These comparisons will be used as base data for reactivity monitoring during subsequent power operation throughout the fuel cycle. At specific power operating conditions, the critical rod configuration will be compared to the configuration expected based upon appropriately corrected past data. This comparison will be made at least every equivalent full power month.

BASES:

3.3 & 4.3 CONTROL ROD SYSTEM

A. Reactivity Limitations

1. Reactivity Margin - Core Loading

The core reactivity limitation is a restriction to be applied principally to the design of new fuel which may be loaded in the core or into a particular refueling pattern. Satisfaction of the limitation can only be demonstrated at the time of loading and must be such that it will apply to the entire subsequent fuel cycle. At each refueling the reactivity of the core loading will be limited so the core can be made subcritical by at least R + 0.25% Δk with the highest worth control rod fully withdrawn and all others inserted. The value of R in % Δk is the amount by which the calculated core reactivity, at any time in the operating cycle, exceeds the reactivity at the time of the demonstration. R must be a positive quantity or zero. The value of R shall include the potential shutdown margin loss assuming full B_4C settling in all inverted poison tubes present in the core. The 0.25% Δk is provided as a finite, demonstrable, sub-criticality margin.

2. Reactivity Margin - Inoperable Control Rods

Specification 3.3.A.2 requires that a rod be taken out of service if it cannot be moved with drive pressure. If a rod is disarmed electrically, its position shall be consistent with the shutdown reactivity limitation stated in Specification 3.3.A.1. This assures that the core can be shutdown at all times with the remaining control rods, assuming the highest worth, operable control rod does rod insert. An allowable pattern for control rods valved out of service will be available to the reactor operator. The number of rods permitted to be inoperable could be many more than the six allowed by the Specification, particularly late in the operation cycle; however, the occurrence of more than six could be indicative of a generic control rod drive problem and the reactor will be shutdown. Also if damage within the control rod drive mechanism and in particular, cracks in drive internal housing, cannot be ruled out, then a generic problem affec ing a number of drives cannot be ruled out. Circumferential cracks resulting from stress assisted intergranular corrosion have occurred in the collet housing of drives at several BWRs. This type of cracking could occur in a number of drives and if the cracks propagated until severance of the collet housing occurred, scram could be prevented in the affected rods. Limiting the period of operation with a potentially severed collet housing and requiring increased surveillance after detecting one stuck rod will assure that the reactor will not be operated with a large number of rods with failed collet housings.

B. Control Rods

1. Control rod dropout accidents as discussed in the FSAR can lead to significant core damage. If coupling integrity is maintained, the possibility of a rod dropout accident is eliminated. The overtravel position feature provides a positive check as only uncoupled drives may reach this position. Neutron instrumentation response to rod movement provides a verification that the rod is following its drive.

BASES: 3.3 & 4.3 (Cont'd)

2. The control rod housing support restricts the outward movement of a control rod to less than 3 inches in the extremely remote event of a housing failure. The amount of reactivity which could be added by this small amount of rod withdrawal, which is less than a normal single withdrawal increment, will not contribute to any damage of the primary coolant system. The design basis is given in Subsection 3.5.2 of the FSAR, and the design evaluation is given in Subsection 3.5.4. This support is not required if the reactor coolant system is at atmospheric pressure since there would then be no driving force to rapidly eject a drive housing.

3. In the course of performing normal startup and shutdown procedures, a pre-specified sequence for the withdrawal or insertion of control rods is followed. Control rod dropout accidents which might lead to significant core damage, cannot occur if this sequence of rod withdrawals or insertions is followed. The Rod Worth Minimizer restricts withdrawals and insertions to those listed in the pre-specified sequence and provides an additional check that the reactor operator is following prescribed sequence. Although beginning a reactor startup without having the RWM operable would entail unnecessary risk, continuing to withdraw rods if the RWM fails subsequently is acceptable if a second licensed operator verifies the withdrawal sequence. Continuing the startup increases core power, reduces the rod worth and reduces the consequences of dropping any rod. Withdrawal of rods for testing is permitted with the RWM inoperable, if the reactor is subcritical and all other rods are fully inserted. Above 20% power, the RWM is not needed since even with a single error an operator cannot withdraw a rod with sufficient worth, which if dropped, would result in anything but minor consequences.

4. Refer to the Vermont Yankee Core Performance Analysis Report.

5. The Source Range Monitor (SRM) system has no scram functions. It does provide the operator with a visual indication of neutron level. The consequences of reactivity accidents are a function of the initial neutron flux. The requirement of at least three counts per second assures that any transient, should it occur, begins at or above the initial value of 10^{-8} of rated power used in the analyses of transients from cold conditions. One operable SRM channel is adequate to monitor the approach to criticality, therefore, two operable SRM's are specified for added conservatism.

6. The Rod Block Monitor (RBM) is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power level operation. During reactor operation with certain limiting control rod patterns, the withdrawal of a designated single control rod could result in one or more fuel rods with MCPR less than the fuel cladding integrity safety limit. During use of such patterns, it is judged that testing of the RBM system prior to withdrawal of such rods will provide added assurance that improper withdrawal does not occur. It is the responsibility of the Nuclear Engineer to identify these limiting patterns and the designated rods either when the patterns are initially established or as they develop due to the occurrence of inoperable control rods.

BASES: 3.3 & 4.3 (Cont'd)

7. Periodic verification that the Scram Discharge Volume (SDV) drain and vent valves are maintained in the open position provides assurance that the SDV will be available to accept the water displaced from the control rod drives in the event of a scram.

C. Scram Insertion Times

The Control Rod System is designed to bring the reactor subcritical at a rate fast enough to prevent fuel damage. The limiting power transient is that resulting from a turbine stop valve closure with a failure of the Turbine Bypass System. Analysis of this transient shows that the negative reactivity rates resulting from the scram with the average response of all the drives as given in the above specification, provide the required protection, and MCPR remains greater than the fuel cladding integrity safety limit.

The scram times for all control rods shall be determined during each operating cycle. The weekly control rod exercise test serves as a periodic check against deterioration of the Control Rod System and also verifies the ability of the control rod drive to scram. The frequency of exercising the control rods under the conditions of two or more control rods valved out of service provides even further assurance of the reliability of the remaining control rods.

D. Control Rod Accumulators

Requiring no more than one inoperable accumulator in any nine-rod (3x3) square array is based on a series of XY PDQ-4 quarter core calculations of a cold, clean core. The worst case in a nine-rod withdrawal sequence resulted in a $K_{eff} \leq 1.0$. Other repeating rod sequences with more rods withdrawn resulted in $K_{eff} \geq 1.0$. At reactor pressures in excess of 800 psig, even those control rods with inoperable accumulators will be able to meet required scram insertion times due to the action of reactor pressure. In addition, they may be normally inserted using the Control-Rod-Drive Hydraulic System. Procedural control will assure that control rods with inoperable accumulators will be spaced in a one-in-nine array rather than grouped together.

E. Reactivity Anomalies

During each fuel cycle, excess operating reactivity varies as fuel depletes and as any burnable poison in supplementary control is burned. The magnitude of this excess reactivity may be inferred from the critical rod configuration. As fuel burnup progresses, anomalous behavior in the excess reactivity may be detected by comparison of the critical rod pattern selected base states to the predicted rod inventory at that state. Power operation base conditions provide the most sensitive and directly interpretable data relative to core reactivity. Furthermore, using power operating base conditions permits frequent reactivity comparisons. Requiring a reactivity comparison at the specified frequency assures that a comparison will be made before the core reactivity change exceeds 1% Δk. Deviations in core reactivity greater than 1% Δk are not expected and require thorough evaluation. One percent reactivity limit is considered safe since an insertion of the reactivity into the core would not lead to transients exceeding design conditions of the Reactor System.

3.4 LIMITING CONDITIONS FOR OPERATION

3.4 REACTOR STANDBY LIQUID CONTROL SYSTEM

Applicability:

Applies to the operating status of the Reactor Standby Liquid Control System.

Objective:

To assure the availability of an independent reactivity control mechanism.

Specification:

A. Normal Operation

Except as specified in 3.4.B below, the Standby Liquid Control System shall be operable during periods when fuel is in the reactor unless:

1. The reactor is in cold shutdown

and

2. Control rods are fully inserted and Specification 3.3.A is met.

4.4 SURVEILLANCE REQUIREMENTS

4.4 REACTOR STANDBY LIQUID CONTROL SYSTEM

Applicability:

Applies to the periodic testing requirement for the Reactor Standby Liquid Control System.

Objective:

To verify the operability of the Standby Liquid Control System.

Specification:

A. Normal Operation

The Standby Liquid Control System shall be verified operable by:

1. Testing pumps and valves in accordance with Specification 4.6.E. A minimum flow rate of 35 gpm at 1275 psig shall be verified for each pump by recirculating demineralized water to the test tank.

2. Verifying the continuity of the explosive charges at least monthly.

In addition, at least once during each operating cycle, the Standby Liquid Control System shall be verified operable by:

1. Testing that the setting of the pressure relief valves is between 1400 and 1490 psig.

2. Initiating one of the standby liquid control loops, excluding the primer chamber and inlet fitting, and verifying that a flow path from a pump to the reactor

3.4 LIMITING CONDITIONS FOR OPERATION

4.4 SURVEILLANCE REQUIREMENTS

vessel is available by pumping demineralized water into the reactor vessel. Both loops shall be tested over the course of two operating cycles.

3. Testing the new trigger assemblies by installing one of the assemblies in the test block and firing it using the installed circuitry. Install the unfired assemblies, taken from the same batch as the fired one, into the explosion valves.

4. Recirculating the borated solution.

3.4 B. Operation with Inoperable Components

From and after the date that a redundant component is made or found to be inoperable, reactor operation is permissible during the succeeding seven days unless such component is sooner made operable.

4.4 B. Operation with Inoperable Components

When a component becomes inoperable, its redundant component shall be or shall have been demonstrated to be operable within 24 hours.

3.4 C. Liquid Poison Tank - Boron Concentration

At all times when the Standby Liquid Control System is required to be operable, the following conditions shall be met:

1. The net volume versus concentration of the sodium pentaborate solution in the standby liquid control tank shall meet the requirements of Figure 3.4.1.

4.4 C. Liquid Poison Tank - Boron Concentration

1. The solution volume in the tank and temperature in the tank and suction piping shall be checked at least daily.

3.4 LIMITING CONDITIONS FOR OPERATION

2. The solution temperature, including that in the pump suction piping, shall be maintained above the curve shown in Figure 3.4.2.

D. If Specification 3.4.A or B is not met, an orderly shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours.

E. If Specification 3.4.C is not met, action shall be immediately initiated to correct the deficiency. If at the end of 12 hours the system has not been restored to full operability, then a shutdown shall be initiated with the reactor in cold shutdown within 24 hours of initial discovery.

4.4 SURVEILLANCE REQUIREMENTS

2. Sodium pentaborate concentration shall be determined at least once a month and within 24 hours following the addition of water or boron, or if the solution temperature drops below the limits specified by Figure 3.4.2.

FIGURE 3.4.1

STANDBY LIQUID CONTROL SYSTEM BORON SOLUTION REQUIREMENTS

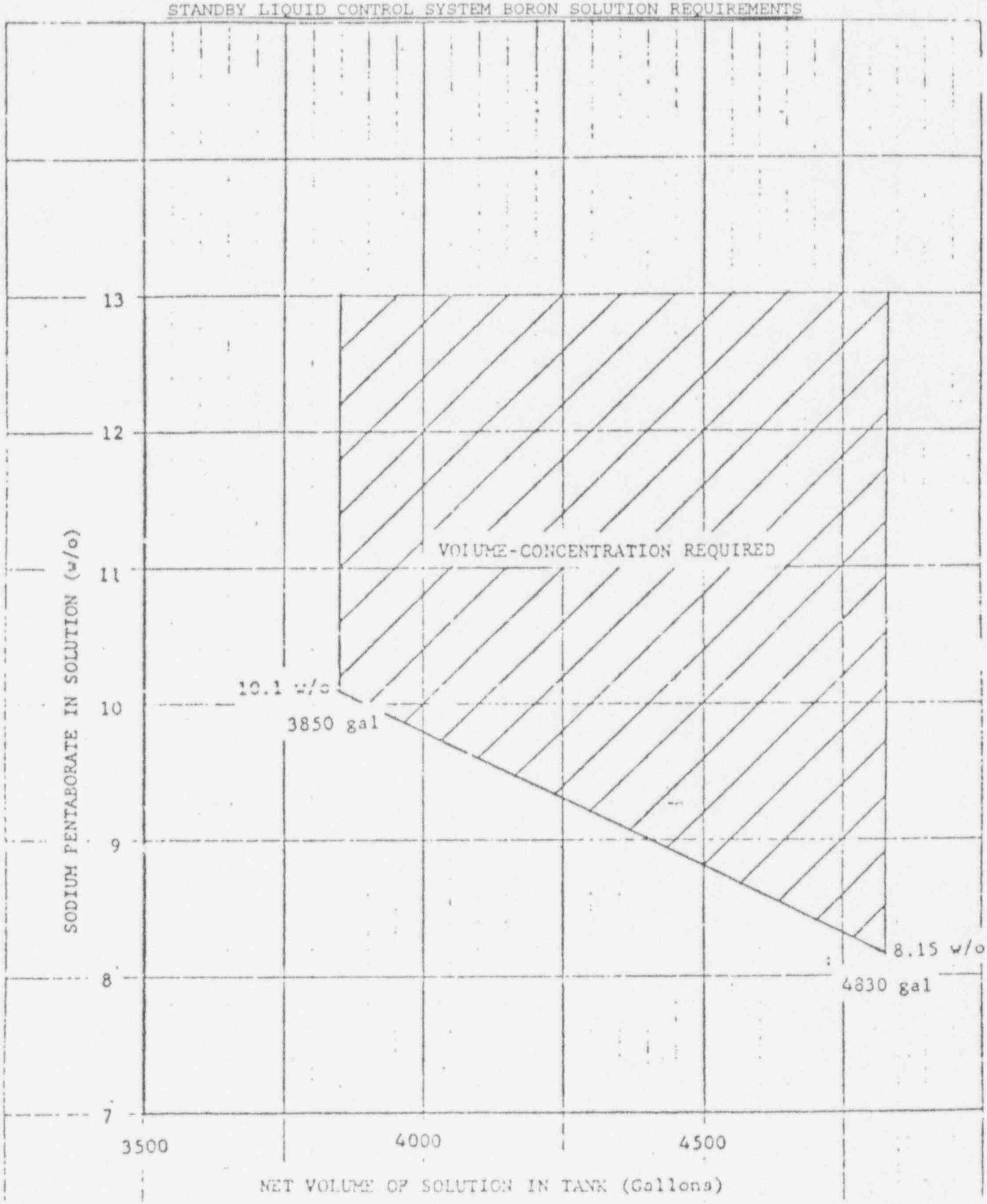
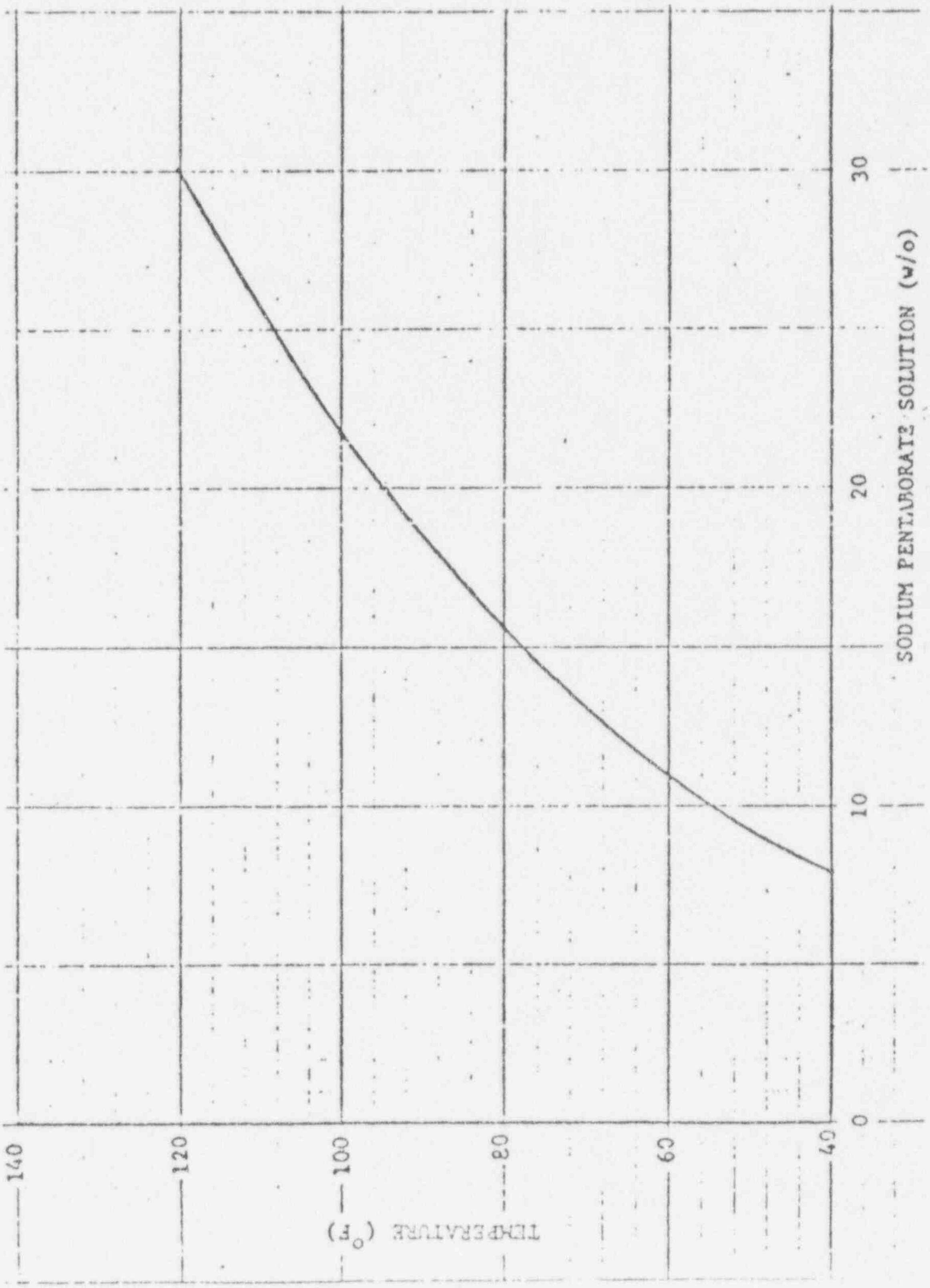


FIGURE 3.4.2

SODIUM PENTABORATE SOLUTION TEMPERATURE REQUIREMENTS



BASES:

3.4 & 4.4 REACTOR STANDBY LIQUID CONTROL SYSTEM

A. Normal Operation

The design objective of the Reactor Standby Liquid Control System is to provide the capability of bringing the reactor from full power to a cold, xenon-free shutdown assuming that none of the withdrawn control rods can be inserted. To meet this objective, the Liquid Control System is designed to inject a quantity of boron which produces a concentration of 800 ppm of boron in the reactor core in less than 138 minutes. An 800 ppm boron concentration in the reactor core is required to bring the reactor from full power to a 5% Δk subcritical condition. An additional margin (25% of boron) is added for possible imperfect mixing of the chemical solution in the reactor water. A minimum quantity of 3850 gallons of solution having a 10.1% sodium pentaborate concentration is required to meet this shutdown requirement.

The time requirement (138 minutes) for insertion of the boron solution was selected to override the rate of reactivity insertion due to cooldown of the reactor following the xenon poison peak. For a required minimum pumping rate of 35 gallons per minute, the maximum net storage volume of the boron solution is established as 4830 gallons.

Boron concentration, solution temperature, and volume are checked on a frequency to assure a high reliability of operation of the system should it ever by required. Pump operability testing in accordance with Specification 4.6.E is adequate to detect if failures have occurred. Flow, relief valve, circuitry, and trigger assembly testing at the prescribed intervals assures a high reliability of system operation capability. Recirculation of the borated solution is done during each operating cycle to ensure one suction line from the boron tank is clear.

B. Operation With Inoperable Components

Only one of the two standby liquid control pumping circuits is needed for proper operation of the system. If one pumping circuit is found to be inoperable, there is no immediate threat to shutdown capability, and reactor operation may continue while repairs are being made. Assurance that the system will perform its intended function is obtained from the results of the pump and valve testing performed in accordance with ASME Section XI requirements. Whenever one redundant component is inoperable, the potential for extended operation with two subsystems inoperable is reduced by requiring that the redundant component be tested within 24 hours.

C. Liquid Poison Tank - Boron Concentration

The solution saturation temperature varies with the concentration of sodium pentaborate. The solution shall be kept at least 10°F above the saturation temperature to guard against boron precipitation. The 10°F margin is included in Figure 3.8.3 of the FSAR. Temperature and liquid level alarms for the system are annunciated in the Control Room.

BASES: 3.4 & 4.4 (Cont'd)

Once the solution has been made up, boron concentration will not vary unless more boron or more water is added. Level indication and alarm indicate whether the solution volume has changed which might indicate a possible solution concentration change. Considering these factors, the test interval has been established.

Sodium pentaborate concentration is determined within 24 hours following the addition of water or boron, or if the solution temperature drops below specified limits. The 24-hour limit allows for 8 hours of mixing, subsequent testing, and notification of shift.

3.5 LIMITING CONDITION FOR OPERATION

3.5 CORE AND CONTAINMENT COOLING SYSTEMS

Applicability:

Applies to the operational status of the Emergency Cooling Subsystems.

Objective:

To assure adequate cooling capability for heat removal in the event of a loss-of-coolant accident or isolation from the normal reactor heat sink.

Specification:

A. Core Spray and Low Pressure Coolant Injection

1. Except as specified in Specifications 3.5.A.2 through 3.5.A.4 below and 3.5.H.3 and 3.5.H.4, both Core Spray and the LPCI Subsystems shall be operable whenever irradiated fuel is in the reactor vessel and prior to a reactor startup from the cold shutdown condition.

4.5 SURVEILLANCE REQUIREMENT

4.5 CORE AND CONTAINMENT COOLING SYSTEMS

Applicability:

Applied to periodic testing of the emergency cooling subsystems.

Objective:

To verify the operability of the core containment cooling subsystems.

Specification:

A. Core Spray and Low Pressure Cooling Injection

Surveillance of the Core Spray and LPCI Subsystems shall be performed as follows:

1. General Testing

Item	Frequency
a. Simulated Automatic Actuation Test	Each refueling outage
b. Operability testing of pumps and valves shall be in accordance with Specification 4.6.E.	
c. Flow Rate Test-Core Spray pumps shall deliver at least 3000 gpm (torus to torus) against a system head of 120 psig. Each LPCI pump shall deliver 7450 ± 150 gpm (vessel to vessel).	Each refueling outage

3.5 LIMITING CONDITION FOR OPERATION

2. From and after the date that one of the Core Spray Subsystems is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days unless such subsystem is sooner made operable, provided that during such seven days, all active components of the other Core Spray Subsystem, the LPCI Subsystems, and the diesel generators required for operation of such components if no external source of power were available, shall be operable.

3. From and after the date that one of the LPCI pumps is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days unless such pump is sooner made operable, provided that during such seven days, the remaining active components of the LPCI Containment Cooling Subsystem and all active components of both Core Spray Subsystems and the diesel generators required for operation of such components if no external source of power were available, shall be operable.

4.5 SURVEILLANCE REQUIREMENT

2. When one Core Spray Subsystem is made or found to be inoperable, the active components of the redundant Core Spray Subsystem shall have been or shall be demonstrated to be operable within 24 hours.

3. When one of the LPCI pumps is made or found to be inoperable, the remaining operable LPCI pumps shall have been or shall be demonstrated to be operable within 24 hours.

3.5 LIMITING CONDITION FOR OPERATION	4.5 SURVEILLANCE REQUIREMENT
4. a. From and after the date that a LPCI Subsystem is made or found to be inoperable due to failure of the associated UPS, reactor operation is permissible only during the succeeding thirty days, for the 1989/90 operating cycle, unless it is sooner made operable, provided that during that time the associated motor control center (89A or 89B) is powered from its respective maintenance tie, all active components of the other LPCI and the Containment Cooling Subsystem, the Core Spray Subsystems, and the emergency diesel generators shall be operable, the requirements of Specification 3.10.A.4 are met, and the 4160 volt tie line to the Vernon Hydro is the operable delayed access power source.	4. When a LPCI Subsystem is made or found to be inoperable, the active components of the redundant LPCI Subsystem shall have been or shall be demonstrated to be operable within 24 hours (except the Recirculation System discharge valves).
b. From and after the date that a LPCI Subsystem is made or found to be inoperable for any reason, other than failure of the UPS during the 1989/90 operating cycle, or Specification 3.5.A.4.a is not met, reactor operation is permissible only during the succeeding seven days unless it is sooner made operable, provided	

3.5 LIMITING CONDITION FOR OPERATION

that during that time all active components of the other LPCI and the Containment Cooling Subsystem, the Core Spray Subsystems, and the diesel generators required for operation of such components if no external source of power were available, shall be operable.

5. All recirculation pump discharge valves and bypass valves shall be operable or closed prior to reactor startup.

6. If the requirements of Specifications 3.5.A cannot be met, an orderly shutdown of the reactor shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours.

B. Containment Spray Cooling Capability

1. Both containment cooling spray loops are required to be operable when the reactor water temperature is greater than 212°F except that a Containment Cooling Subsystem may be inoperable for thirty days.

2. If this requirement cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours.

4.5 SURVEILLANCE REQUIREMENT

5. Recirculation pump discharge valves shall be tested to verify full open to full closed in $27 \leq t \leq 33$ seconds and bypass valves shall be tested for operability in accordance with Specification 4.6.E.

B. Containment Spray Cooling Capability

1. Surveillance of the drywell spray loops shall be performed as follows. During each five-year period, an air test shall be performed on the drywell spray headers and nozzles.

2. When a Containment Cooling Subsystem is made or found to be inoperable, the active components of the redundant Containment Cooling Subsystem shall have been or shall be demonstrated to be operable within 24 hours.

3.5 LIMITING CONDITION FOR OPERATION

C. Residual Heat Removal (RHR) Service Water System

1. Except as specified in Specifications 3.5.C.2, and 3.5.C.3 below, both RHR Service Water Subsystem loops shall be operable whenever irradiated fuel is in the reactor vessel and prior to reactor startup from a cold condition.

2. From and after the date that one of the RHR service water pumps is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding thirty days unless such pump is sooner made operable, provided that during such thirty days all other active components of the RHR Service Water Subsystem are operable.

3. From and after the date that one RHR Service Water Subsystem is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days unless such subsystem is sooner made operable, provided that all active components of the other RHR Service Water

4.5 SURVEILLANCE REQUIREMENT

C. Residual Heat Removal (RHR) Service Water System

Surveillance of the RHR Service Water System shall be performed as follows:

1. RHR Service Water Subsystem testing:

   Operability testing of pumps and valves shall be in accordance with Specification 4.6.E.

   Each RHR service water pump shall deliver at least 2700 gpm and a pressure of at least 70 psia shall be maintained at the RHR heat exchanger service water outlet when the corresponding pairs of RHR service water pumps and station service water pumps are operating.

2. When one of the RHR service water pumps is made or found to be inoperable, the operable RHR service water pumps shall have been or shall be demonstrated to be operable within 24 hours.

3. When one RHR Service Water Subsystem is made or found to be inoperable, the active components of the redundant RHR Service Water Subsystem shall have been or shall be demonstrated to be operable within 24 hours.

3.5 LIMITING CONDITION FOR OPERATION

Subsystem, both Core Spray Subsystems, and both diesel generators required for operation of such components if no external source of power were available, shall be operable.

4. If the requirements of Specification 3.5.C cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours.

D. Station Service Water and Alternate Cooling Tower Systems

1. Except as specified in Specifications 3.5.D.2 and 3.5.D.3, both Station Service Water Subsystem loops and the alternate cooling tower shall be operable whenever irradiated fuel is in the reactor vessel and reactor coolant temperature is greater than 212°F.

2. From and after the date that one of the Station Service Water Subsystems is made or found inoperable for any reason, reactor operation is permissible only during the succeeding 15 days unless such subsystem is made operable, provided that during such 15 days all other active components of the Station Service Water and Alternate Cooling Tower Systems are operable.

4.5 SURVEILLANCE REQUIREMENT

D. Station Service Water and Alternate Cooling Tower Systems

Surveillance of the Station Service Water and Alternate Cooling Tower Systems shall be performed as follows:

1. Operability testing of pumps and valves shall be in accordance with Specification 4.6.E. Each pump shall deliver at least 2700 gpm against a TDH of 250 feet.

2. When one Station Service Water Subsystem is made or found to be inoperable, the active components of the redundant Station Service Water Subsystem and the alternate cooling tower fan shall have been or shall be demonstrated to be operable within 24 hours.

3.5 LIMITING CONDITION FOR OPERATION

3. From and after the date that the Alternate Cooling Tower Subsystem or both Station Service Water Subsystems are made or found inoperable for any reason, reactor operation is permissible only during the succeeding seven days unless such subsystems are made operable, provided that during such seven days all other active components of the other subsystem are operable.

4. If the requirements of Specification 3.5.D cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours.

E. <u>High Pressure Cooling Injection (HPCI) System</u>

1. Except as specified in Specification 3.5.E.2, whenever irradiated fuel is in the reactor vessel and reactor pressure is greater than 150 psig and prior to reactor startup from a cold condition:

   a. The HPCI System shall be operable.

   b. The condensate storage tank shall contain at least 75,000 gallons of condensate water.

4.5 SURVEILLANCE REQUIREMENT

3. When the Alternate Cooling Subsystem or both Station Service Water Subsystems are made or found to be inoperable, the operable subsystem shall have been or shall be demonstrated to be operable within 24 hours.

E. <u>High Pressure Coolant Injection (HPCI) System</u>

Surveillance of HPCI System shall be performed as follows:

1. <u>Testing</u>

Item	Frequency
Simulated Automatic Actuation Test	Each refueling outage

Operability testing of the pump and valves shall be in accordance with Specification 4.6.E. The HPCI System shall deliver at least 4250 gpm at normal reactor operating pressure when recirculating to the Condensate Storage Tank.

3.5 LIMITING CONDITION FOR OPERATION

2. From and after the date that the HPCI Subsystem is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days unless such subsystem is sooner made operable, provided that during such seven days all active components of the Automatic Depressurization Subsystems, the Core Spray Subsystems, the LPCI Subsystems, and the RCIC System are operable.

3. If the requirements of Specification 3.5.E cannot be met, an orderly shutdown shall be initiated and the reactor pressure shall be reduced to 120 psig within 24 hours.

F. Automatic Depressurization System

1. Except as specified in Specification 3.5.F.2 below, the entire Automatic Depressurization Relief System shall be operable at any time the reactor pressure is above 100 psig and irradiated fuel is in the reactor vessel.

2. From and after the date that one of the four relief valves of the Automatic Depressurization Subsystem are made or found to be inoperable due to malfunction of the electrical portion of the valve when the

4.5 SURVEILLANCE REQUIREMENT

2. When the HPCI Subsystem is made or found to be inoperable, the Automatic Depressurization System shall have been or shall be demonstrated to be operable within 24 hours.

NOTE: Automatic Depressurization System operability shall be demonstrated by performing a functional test of the trip system logic.

F. Automatic Depressurization System

Surveillance of the Automatic Depressurization System shall be performed as follows:

1. Operability testing of the relief valves shall be in accordance with Specification 4.6.E.

2. When one relief valve of the Automatic Pressure Relief Subsystem is made or found to be inoperable, the HPCI Subsystem shall have been or shall be demonstrated to be operable within 24 hours.

3.5 LIMITING CONDITION FOR OPERATION

reactor is pressurized above 100 psig with irradiated fuel in the reactor vessel, continued reactor operation is permissible only during the succeeding seven days unless such a valve is sooner made operable, provided that during such seven days both the remaining Automatic Relief System valves and the HPCI System are operable.

3. If the requirements of Specification 3.5.F cannot be met, an orderly shutdown shall be initiated and the reactor pressure shall be reduced to 100 psig within 24 hours.

G. Reactor Core Isolation Cooling System (RCIC)

1. Except as specified in Specification 3.5.G.2 below, the RCIC System shall be operable whenever the reactor pressure is greater than 150 psig and irradiated fuel is in the reactor vessel.

2. From and after the date that the RCIC System is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding 7 days unless such system is sooner made operable, provided that during such 7 days all active components of the HPCI System are operable.

4.5 SURVEILLANCE REQUIREMENT

G. Reactor Core Isolation Cooling System (RCIC)

Surveillance of the RCIC System shall be performed as follows:

1. Testing

Item	Frequency
Simulated automatic actuation test (testing valve operability)	Each refueling outage

Operability testing of the pump and valves shall be in accordance with Specification 4.6.E. The RCIC System shall deliver at least 400 gpm at normal operating pressure when recirculating to the Condensate Storage Tank.

3.5 LIMITING CONDITION FOR OPERATION	4.5 SURVEILLANCE REQUIREMENT

3. If the requirements of Specification 3.5.G cannot be met, an orderly shutdown shall be initiated and the reactor pressure shall be reduced to 120 psig within 24 hours.

H. Minimum Core and Containment Cooling System Availability

1. During any period when one of the standby diesel generators is inoperable, continued reactor operation is permissible only during the succeeding seven days, provided that all of the Low Pressure Core Cooling and Containment Cooling Subsystems connecting to the operable diesel generator shall be operable. If this requirement cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours.

2. Any combination of inoperable components in the Core and Containment Cooling Systems shall not defeat the capability of the remaining operable components to fulfill the core and containment cooling functions.

3. When irradiated fuel is in the reactor vessel and the reactor is in the cold shutdown condition, all Core and Containment Cooling Subsystems may be inoperable provided no work is permitted which has the potential for draining the reactor vessel.

H. Minimum Core and Containment Cooling System Availability

1. When one of the standby diesel generators is made or found to be inoperable, the remaining diesel generator shall have been or shall be demonstrated to be operable within 24 hours.

3.5 LIMITING CONDITION FOR OPERATION

4. When irradiated fuel is in the reactor vessel and the reactor is in the refueling condition, both LPCI subsystems, or both Core Spray systems, or one diesel generator may be inoperable provided that a source of water of greater than 300,000 gal. is available to the operable core cooling subsystem.

I. Maintenance of Filled Discharge Pipe

Whenever core spray subsystems, LPCI subsystem, HPCI, or RCIC are required to be operable, the discharge piping from the pump discharge of these systems to the last block valve shall be filled.

4.5 SURVEILLANCE REQUIREMENT

I. Maintenance of Filled Discharge Pipe

The following surveillance requirements shall be adhered to to assure that the discharge piping of the core spray subsystems, LPCI subsystem, HPCI and RCIC are filled:

1. Every month and prior to the testing of the LPCI subsystem and core spray subsystem, the discharge piping of these systems shall be vented from the high point and water flow observed.

2. Following any period where the LPCI subsystem or core spray subsystems have not been required to be operable, the discharge piping of the inoperable system shall be vented from the high point prior to the return of the system to service.

3. Whenever the HPCI or RCIC system is lined up to take suction from the torus, the discharge piping of the HPCI and RCIC shall be vented from the high point of the system and water flow observed on a monthly basis.

BASES:

3.5 CORE AND CONTAINMENT COOLANT SYSTEMS

A. Core Spray Cooling System and Low Pressure Coolant Injection System

This Specification assures that adequate standby cooling capability is available whenever irradiated fuel is in the Reactor Vessel.

Based on the loss-of-coolant analyses, the Core Spray and LPCI Systems provide sufficient cooling to the core to dissipate the energy associated with the loss-of-coolant accident and to limit the accident-caused core conditions as specified in 10CFR50, Appendix K. The analyses consider appropriate combinations of the two Core Spray Subsystems and the two LPCI Subsystems associated with various break locations and equipment availability in accordance with required single failure assumptions. (Each LPCI Subsystem consists of the LPCI pumps, the recirculation pump discharge valve, and the LPCI injection valve which combine to inject torus water into a recirculation loop.)

The LPCI System is designed to provide emergency cooling to the core by flooding in the event of a loss-of-coolant accident. This system is completely independent of the Core Spray System; however, it does function in combination with the Core Spray System to prevent excessive fuel clad temperature. The LPCI and the Core Spray Systems provide adequate cooling for break areas up to and including the double-ended recirculation line break without assistance from the high pressure emergency Core Cooling Subsystems.

The intent of these specifications is to prevent startup from the cold condition without all associated equipment being operable. However, during operation, certain components may be out of service for the specified allowable repair times. Assurance that the systems will perform their intended function is obtained from the results of the pump and valve testing performed in accordance with ASME Section XI requirements referenced in Specification 4.6.E. Whenever one redundant system is inoperable, the potential for extended operation with two subsystems inoperable is reduced by requiring that the redundant subsystem be tested within 24 hours.

B. and C. Containment Spray Cooling Capability and RHR Service Water System

The containment heat removal portion of the RHR System is provided to remove heat energy from the containment in the event of a loss-of-coolant accident. For the flow specified, the containment long-term pressure is limited to less than 5 psig and, therefore, the flow is more than ample to provide the required heat removal capability. Reference: Section 14.6.3.3.2 FSAR.

Each Containment Cooling Subsystem consists of two RHR service water pumps, 1 heat exchanger, and 2 RHR (LPCI) pumps. Either set of equipment is capable of performing the containment cooling function. In fact, an analysis in Section 14.6 of the FSAR shows that one subsystem consisting of 1 RHR service water pump, 1 heat exchanger, and 1 RHR pump has sufficient capacity to perform the cooling function. Assurance that the systems will perform their intended function is obtained from the results of the pump and valve testing performed in accordance with ASME Section XI requirements referenced in Specification 4.6.E. Whenever one redundant system is inoperable, the potential for extended operation with two subsystems inoperable is reduced by requiring that the redundant subsystem be tested within 24 hours.

BASES: 3.5 (Cont'd)

D. Station Service Water and Alternate Cooling Tower System

The Station Service Water Subsystems and the Alternate Cooling Tower System provide alternate heat sinks to dissipate residual heat after a shutdown or accident. Each Station Service Water Subsystem and the Alternate Cooling Tower System provides sufficient heat sink capacity to perform the required heat dissipation. The Alternate Cooling Tower System will provide the necessary heat sink in the event both Station Service Water Subsystems become incapacitated due to a loss of the Vernon Dam with subsequent loss of the Vernon Pond.

E. High Pressure Coolant Injection System

The High Pressure Coolant Injection System (HPCIs) is provided to adequately cool the core for all pipe breaks smaller than those for which the LPCI or Core Spray Cooling Subsystems can protect the core.

The HPCIs meets this requirement without the use of outside power. For the pipe breaks for which the HPCIs is intended to function the core never uncovers and is continuously cooled; thus, no clad damage occurs and clad temperatures remain near normal throughout the transient. Reference: Subsection 6.5.2.2 of the FSAR.

F. Automatic Depressurization System

The relief valves of the Automatic Depressurization System are a backup to the HPCIs. They enable the Core Spray Cooling System or LPCI Subsystem to provide protection against the small pipe break in the event of HPCI failure by depressurizing the reactor vessel rapidly enough to actuate the Core Sprays or LPCI Subsystem. Either of the two Core Spray Cooling Systems or LPCIs provides sufficient flow of coolant to prevent clad melting. All four relief valves are included in the Automatic Pressure Relief System. (See VYNPS, FSAR Vol. 4, Appendix B.)

G. Reactor Core Isolation Cooling System

The Reactor Core Isolation Cooling System (RCIC) is provided to maintain the water inventory of the reactor vessel in the event of a main steam line isolation and complete loss of outside power without the use of the emergency core cooling systems. The RCIC meets this requirement. Reference Section 14.5.4.4 FSAR. The HPCIS provides an incidental backup to the RCIC system such that in the event the RCIC should be inoperable no loss of function would occur if the HPCIS is operable.

H. Minimum Core and Containment Cooling System Availability

The core cooling and the containment cooling subsystems provide a method of transferring the residual heat following a shutdown or accident to a heat sink. Based on analyses, this specification assures that adequate cooling capacity is available by precluding any combination of inoperable components from fulfilling the core and containment cooling function. It is permissible, based upon the low heat load and other methods available to remove the residual heat, to disable all core and containment cooling systems for maintenance if the reactor is cold and shutdown and there is no potential for draining the reactor vessel. However, if refueling operations are in progress, one coolant injection system, one diesel and a residual of at least 300,000 gallons is required to assure core flooding capability.

BASES: 3.5 (Cont'd)

I. Maintenance of Filled Discharge Pipe

Full discharge lines are required when the core spray subsystems, HPCI and RCIC are required to be operable to preclude the possibility of damage to the discharge piping due to water hammer action upon a pump start.

BASES:

4.5 CORE AND CONTAINMENT COOLANT SYSTEMS

A. Core Spray and LPCI

During normal plant operation, manual tests of operable pumps and valves shall be conducted in accordance with Specification 4.6.E to demonstrate operability.

During each refueling shutdown, tests (as summarized below) shall be conducted to demonstrate proper automatic operation and system performance.

Periodic testing as described in Specification 4.6.E will demonstrate that all components which do not operate during normal conditions will operate properly if required.

The automatic actuation test will be performed by simulation of high drywell pressure or low-low water level. The starting of the pump and actuation of valves will be checked. The normal power supply will be used during the test. Testing of the sequencing of the pumps when the diesel generator is the source of power will be checked during the testing of the diesel. Following the automatic actuation test, the flow rate will be checked by recirculation to the suppression chamber. The pump and valve operability checks will be performed by manually starting the pump or activating the valve. For the pumps, the pump motors will be run long enough for them to reach operating temperatures.

B. and C. Containment Spray Cooling Capability and RHR Service Water Systems

The periodic testing requirements specified in Specifications 4.5.B and C will demonstrate that all components will operate properly if required. Since this is a manually actuated system, no automatic actuation test is required. The system will be activated manually and the flow checked by an indicator in the control room.

Once every five years air tests will be performed to assure that the containment spray header nozzles are operable.

D., E., and F. Station Service Water and Alternate Cooling Tower Systems and High Pressure Coolant Injection and Automatic Depressurization System

The testing described in Specification 4.6.E for the HPCI System will demonstrate that the system will operate if required. The Automatic Depressurization System is tested during refueling outages to avoid an undesirable blowdown of the Reactor Coolant System.

The HPCI Automatic Actuation Test will be performed by simulation of the accident signal. The test is normally performed in conjunction with the automatic actuation of all Core Standby Cooling Systems.

G. Reactor Core Isolation Cooling System

Frequency of testing of the RCIC System is the same as the HPCI, per Specification 4.6.E, and demonstrates that the system is operable if needed.

BASES: 4.5 (Cont'd)

H. Minimum Core and Containment Cooling System Availability

Assurance that the diesels will perform their intended function is obtained by the periodic surveillance test and the results obtained from the pump and valve testing performed in accordance with ASME Section XI requirements described in Specification 4.6.E. Whenever a diesel is inoperable, the potential for extended operation with two diesels inoperable is reduced by requiring that the redundant diesel be tested within 24 hours.

I. Maintenance of Filled Discharge Pipe

Observation of water flowing from the discharge line high point vent as discussed in Section 1 assures that the Core Cooling Subsystems will not experience water hammer damage when any of the pumps are started. Core Spray Subsystems and LPCI Subsystems will also be vented through the discharge line high point vent following a return from an inoperable status to assure that the system is "solid" and ready for operation.

3.6 LIMITING CONDITIONS FOR OPERATION

3.6 REACTOR COOLANT SYSTEM

Applicability:

Applies to the operating status of the reactor coolant system.

Objective:

To assure the integrity and safe operation of the reactor coolant system.

Specification:

A. Pressure and Temperature Limitations

1. The reactor coolant system temperature and pressure shall be limited in accordance with the limit lines shown on Figure 3.6.1 during heatup, cooldown, criticality (except for the purposes of low power physics testing), and inservice leak and hydrostatic testing.

2. The maximum heatup or cooldown rate is 100°F when averaged over any one hour period.

3. The reactor vessel head bolting shall not be tensioned unless the temperature of the vessel head flange and the head is greater than 70°F.

4. The pump in an idle recirculation loop shall not be started unless the temperatures of the coolant within the idle and operating recirculation loops are within 50°F of each other.

4.6 SURVEILLANCE REQUIREMENTS

4.6 REACTOR COOLANT SYSTEM

Applicability:

Applies to the periodic examination and testing requirements for the reactor coolant system.

Objective:

To determine the condition of the reactor coolant system and the operation of the safety devices related to it.

Specification:

A. Pressure and Temperature Limitations

1. The reactor coolant temperature and pressure shall be recorded at least once per hour during system heatup, cooldown and inservice leak and hydrostatic testing operations.

2. The reactor coolant temperature and pressure shall be recorded at the time of reactor criticality.

3. When the reactor vessel head bolting is being tightened or loosened, the reactor vessel shell temperature immediately below the vessel flange shall be permanently recorded.

4. Prior to and after startup of an idle recirculation loop, the temperature of the reactor coolant in the operating and idle loops shall be recorded.

3.6 LIMITING CONDITIONS FOR OPERATION

4.6 SURVEILLANCE REQUIREMENTS

5. The reactor vessel irradiation surveillance specimens shall be removed and examined to determine changes in material properties in accordance with the following schedule:

CAPSULE	REMOVAL YEAR
1	10
2	30
3	Standby

The results shall be used to update Figures 3.6.2 and 3.6.3. The removal times shall be referenced to the refueling outage following the year specified, referenced to the date of commercial operation.

3.6 B. Coolant Chemistry

1. a. During reactor power operation, the radioiodine concentration in the reactor coolant shall not exceed 1.1 microcuries of I-131 dose equivalent per gram of water, except as allowed in Specification 3.6.B.1.b.

4.6 B. Coolant Chemistry

1. a. A sample of reactor coolant shall be taken at least every 96 hours and analyzed for radioactive iodines of I-131 through I-135 during power operation. In addition, when steam jet air ejector monitors indicate an increase in radioactive gaseous effluents of 25 percent or 5000 µCi/sec, whichever is greater, during steady state reactor operation a reactor coolant sample shall be taken and analyzed for radioactive iodines.

3.6 LIMITING CONDITIONS FOR OPERATION	4.6 SURVEILLANCE REQUIREMENTS
b. The radioiodine concentration in the reactor coolant shall not exceed 1.1 microcuries of I-131 dose equivalent per gram of water, for greater than 24 consecutive hours.	b. An isotopic analysis of a reactor coolant sample shall be made at least once per month.
c. The radioiodine concentration in the reactor coolant shall not exceed 4.0 microcuries of I-131 dose equivalent per gram of water.	c. Whenever the radioiodine concentration of prior steady-state reactor operation is greater than 0.011 μCi/gm but less than 0.11 μCi/gm, a sample of reactor coolant shall be taken within 24 hours of the next reactor startup and analyzed for radioactive iodines of I-131 through I-135.
	d. Whenever the radioiodine concentration of prior steady-state reactor operation is greater than 0.11 μCi/gm, a sample of reactor coolant shall be taken prior to the next reactor startup and analyzed for radioactive iodines of I-131 through I-135, as well as within 24 hours following a reactor startup.

3.6 LIMITING CONDITIONS FOR OPERATION

4.6 SURVEILLANCE REQUIREMENTS

e. With the radioiodine concentration in the reactor coolant greater than 1.1 microcuries/ gram dose equivalent I-131, a sample of reactor coolant shall be taken every 4 hours and analyzed for radioactive iodines of I-131 through I-135, until the specific activity of the reactor coolant is restored below 1.1 microcuries/ gram dose equivalent I-131.

2. The reactor coolant water shall not exceed the following limits with steaming rates less than 100,000 pounds per hour except as specified in Specification 3.6.B.3:

Conductivity	5umho/cm
Chloride ion	0.01 ppm

2. During startups and at steaming rates below 100,000 pounds per hour, a sample of reactor coolant shall be taken every four hours and analyzed for conductivity and chloride content.

3. For reactor startups the maximum value for conductivity shall not exceed 10 umho/cm and the maximum value for chloride ion concentration shall not exceed 0.1 ppm, in the reactor coolant water for the first 24 hours after placing the reactor in the power operating condition.

3. a. With steaming rates greater than or equal to 100,000 pounds per hour, a reactor coolant sample shall be taken at least every 96 hours and when the continuous conductivity monitors indicate abnormal conductivity (other than short-term spikes), and analyzed for conductivity and chloride ion content.

3.6 LIMITING CONDITIONS FOR OPERATION

4. Except as specified in Specification 3.6.B.3 above, the reactor coolant water shall not exceed the following limits with steaming rates greater than or equal to 100,000 pounds per hours.

Conductivity	5 uhmo/cm
Chloride ion	0.5 ppm

5. If Specification 3.6.B is not met, an orderly shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours.

C. Coolant Leakage

1. Any time irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212°F, reactor coolant leakage into the primary containment from unidentified sources shall not exceed 5 gpm. In addition, the total reactor coolant system leakage into the primary containment shall not exceed 25 gpm.

2. Both the sump and air sampling systems shall be operable during power operation. From and after the date that one of these systems is made or found inoperable for any reason, reactor operation is permissible only during succeeding seven days.

4.6 SURVEILLANCE REQUIREMENTS

b. When the continuous conductivity monitor is inoperable, a reactor coolant sample shall be taken every four hours and analyzed for conductivity and chloride ion content.

C. Coolant Leakage

Reactor coolant system leakage shall be checked and logged at least once per day.

3.6 LIMITING CONDITIONS FOR OPERATION

3. If these conditions cannot be met, initiate an orderly shutdown and the reactor shall be in the cold shutdown condition within 24 hours.

D. Safety and Relief Valves

1. During reactor power operating conditions and whenever the reactor coolant pressure is greater than 120 psig and temperature greater than 350°F, both safety valves shall be operable. The relief valves shall be operable, except that if one relief valve is inoperable, reactor power shall be immediately reduced to and maintained at or below 95% of rated power.

2. If Specification 3.6.D.1 is not met, initiate an orderly shutdown and the reactor coolant pressure shall be below 120 psig and 350°F within 24 hours.

E. Structural Integrity and Operability Testing

The structural integrity and the operability of the safety-related systems and components shall be maintained at the level required by the original acceptance standards throughout the life of the plant.

4.6 SURVEILLANCE REQUIREMENTS

D. Safety and Relief Valves

1. Operability testing of Safety and Relief Valves shall be in accordance with Specification 4.6.E. The lift point of the safety and relief valves shall be set as specified in Specification 2.2.B.

E. Structural Integrity and Operability Testing

1. Inservice inspection of safety-related components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the NRC pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).

3.6 LIMITING CONDITIONS FOR OPERATION

4.6 SURVEILLANCE REQUIREMENTS

2. Operability testing of safety-related pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the NRC pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).

F. Jet Pumps

1. Whenever the reactor is in the startup/hot standby or run modes, all jet pumps shall be intact and all operating jet pumps shall be operable. If it is determined that a jet pump is inoperable, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours.

2. Flow indication from each of the twenty jet pumps shall be verified prior to initiation of reactor startup from a cold shutdown condition.

F. Jet Pumps

1. Whenever there is recirculation flow with the reactor in the startup/hot standby or run modes, jet pump integrity and operability shall be checked daily by verifying that the following two conditions do not occur simultaneously:

   a. The recirculation pump flow differs by more than 10% from the established speed-flow characteristics.

   b. The indicated total core flow is more than 10% greater than the core flow value derived from established power-core flow relationships.

2. Additionally, when operating with one recirculation pump, the diffuser to lower plenum differential pressure shall be checked daily, and the differential pressure of any jet pump in the idle loop shall not vary by more than 10% from established patterns.

3.6 LIMITING CONDITIONS FOR OPERATION

3. The indicated core flow is the sum of the flow indication from each of the twenty jet pumps. If flow indication failure occurs for two or more jet pumps, immediate corrective action shall be taken. If flow indication for all but one jet pump cannot be obtained within 12 hours an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours.

G. Single Loop Operation

1. The reactor may be started and operated or operation may continue with a single recirculation loop provided that:

   a. The designated adjustments for APRM flux scram and rod block trip settings (Specifications 2.1.A.1.a and 2.1.B.1, Table 3.1.1 and Table 3.2.5), rod block monitor trip setting (Table 3.2.5), MCPR fuel cladding integrity safety limit (Specification 1.1.A), and MCPR operating limits and MAPLHGR limits, provided in the Core Operating Limits Report, are initiated within 8 hours. During the next 12 hours, either these adjustments must be completed or the reactor brought to Hot Shutdown.

4.6 SURVEILLANCE REQUIREMENTS

3. The baseline data required to evaluate the conditions in Specifications 4.6.F.1 and 4.6.F.2 shall be acquired each operating cycle.

G. Single Loop Operation

1. With one recirculation pump not in operation, core flow between 34% and 45% of rated, and core thermal power greater than the limit specified in Figure 3.6.4 (Region 2), establish baseline APRM and LPRM(1) neutron flux noise levels prior to entering this region, provided that baseline values have not been established since the last core refueling. Baseline values shall be established with one recirculation pump not in operation and core thermal power less than or equal to the limit specified in Figure 3.6.4.

(1) Detector Levels A and C of one LPRM string per core octant plus detector Levels A and C of one LPRM string in the center of the core shall be monitored.

3.6 LIMITING CONDITIONS FOR OPERATION	4.6 SURVEILLANCE REQUIREMENTS
b. With one recirculation pump not in operation, core thermal power greater than the limit specified in Figure 3.6.4, and core flow between 34% and 45% of rated (Region 2 of Figure 3.6.4): (1) If baseline APRM and LPRM[(1)] neutron flux noise levels have been established since the last core refueling, initiate action within 15 minutes such that the APRM and LPRM[(1)] neutron flux noise levels are determined within 2 hours, and: (i) If the APRM and LPRM[(1)] neutron flux noise levels are less than or equal to 3 times their established baseline levels, continue to determine the noise levels at least once per 8 hours and within 30 minutes after the completion	

(1) Detector Levels A and C of one LPRM string per core octant plus detector Levels A and C of one LPRM string in the center of the core shall be monitored.

3.6 LIMITING CONDITIONS FOR OPERATION	4.6 SURVEILLANCE REQUIREMENTS
of a core thermal power increase greater than 5% of rated core thermal power, or (ii) If the APRM and/or LPRM[1] neutron flux noise levels are greater than 3 times their established baseline levels, initiate action within 15 minutes such that the noise levels are restored to within the required limits within 2 hours by increasing core flow and/or by initiating an orderly reduction of core thermal power by inserting control rods.	

(1) Detector Levels A and C of one LPRM string per core octant plus detector Levels A and C of one LPRM string in the center of the core shall be monitored.

3.6 LIMITING CONDITIONS FOR OPERATION

(2) If baseline APRM and LPRM[1] neutron flux noise levels have not been established since the last core refueling, initiate action within 15 minutes of entering this region (Region 2 of Figure 3.6.4) such that operation is outside this region within 2 hours.

c. With one recirculation pump not in operation, core thermal power greater than the limit specified in Figure 3.6.4, and core flow less than 34% of rated (Region 1 of Figure 3.6.4), initiate action within 15 minutes of entering this region such that operation is outside this region within 2 hours.

d. The idle loop is isolated by electrically disarming the breaker to the recirculation pump motor generator set drive motor prior to startup or, if disabled during reactor operation, within 24 hours, and until such time as the inactive recirculation loop is to be returned to service.

4.6 SURVEILLANCE REQUIREMENTS

(1) Detector Levels A and C of one LPRM string per core octant plus detector Levels A and C of one LPRM string in the center of the core shall be monitored.

3.6 LIMITING CONDITIONS FOR OPERATION

e. The recirculation system controls will be placed in the manual flow control mode.

H. <u>Recirculation System</u>

1. With two recirculation pumps in operation, with total core flow less than 45% of rated, and core thermal power greater than the limit specified in Figure 3.6.4 (Regions 1 and 2):

a. If baseline APRM and LPRM[(1)] neutron flux noise levels have been established since the last core refueling, initiate action within 15 minutes such that the APRM and LPRM[(1)] neutron flux noise levels are determined within 2 hours, and:

(1) If the APRM and LPRM[(1)] neutron flux noise levels are less than or equal to 3 times their established baseline levels, continue to determine the noise levels at least once per 8 hours and within 30 minutes after the completion of a core thermal

4.6 SURVEILLANCE REQUIREMENTS

H. <u>Recirculation System</u>

1. With two recirculation pumps in operation, total core flow less than 45% of rated, and core thermal power greater than the limit specified in Figure 3.6.4 (Regions 1 and 2), establish baseline APRM and LPRM[(1)] neutron flux noise levels prior to entering these regions, provided that baseline values have not been established since the last core refueling. Baseline values shall be established with core thermal power less than or equal to the limit specified in Figure 3.6.4.

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(1) Detector Levels A and C of one LPRM string per core octant plus detector Levels A and C of one LPRM string in the center of the core shall be monitored.

3.6 LIMITING CONDITIONS FOR OPERATION	4.6 SURVEILLANCE REQUIREMENTS
power increase greater than 5% of rated core thermal power, or (2) If the APRM and/or LPRM[(1)] neutron flux noise levels are greater than 3 times their established baseline levels, initiate action within 15 minutes such that the noise levels are restored to within the required limits within 2 hours by increasing core flow and/or by initiating an orderly reduction of core thermal power by inserting control rods. b. If baseline APRM and LPRM[(1)] neutron flux noise levels have not been established since the last core refueling, initiate action within 15 minutes of entering these regions (Regions 1 and 2 of Figure 3.6.4) such that operation is outside these regions within 2 hours.	

(1) Detector Levels A and C of one LPRM string per core octant plus detector Levels A and C of one LPRM string in the center of the core shall be monitored.

3.6 LIMITING CONDITIONS FOR OPERATION

2. Operation with one recirculation loop is permitted according to Specification 3.6.G.1.

3. With no reactor coolant system recirculation loops in operation, immediately initiate an orderly reduction in core thermal power to less than or equal to the limit specified in Figure 3.6.4 (Region 3), and initiate measures such that the unit is in Hot Shutdown within the next 12 hours.

I. Shock Suppressors (Snubbers)

1. Except as noted in 3.6.I.2 and 3.6.I.3 below, all required safety-related snubbers shall be operable whenever its supported system is required to be operable.

2. With one or more required snubbers inoperable, within 72 hours, replace or restore the snubber to operable status and perform an engineering evaluation per Specification 4.6.I.1b and c, on the supported component. In all cases, the required snubbers shall be made operable or replaced prior to reactor startup.

3. If the requirements of 3.6.I.1 and 3.6.I.2 cannot be met, the supported system shall be declared inoperable and the appropriate action statement for that system shall be followed.

4.6 SURVEILLANCE REQUIREMENTS

I. Shock Suppressors (Snubbers)

1. Each snubber shall be demonstrated operable by performance of the following inspection program.

   a. Visual Inspections

      Visual inspections shall be performed in accordance with the following schedule:

No. Inoperable Snubbers Per Inspection Period	Next Required Inspection Intervals
0	18 months ±25%
1	12 months ±25%
2	6 months ±25%
3, 4	124 days ±25%
5, 6, 7	62 days ±25%
8 or more	31 days ±25%

      The snubbers may be categorized into two groups: the accessible and those inaccessible during reactor operation. Each group may be inspected independently in accordance with the above schedule.

3.6 LIMITING CONDITIONS FOR OPERATION	4.6 SURVEILLANCE REQUIREMENTS

The inspection interval shall not be lengthened more than one step at a time. Inaccessible snubbers are required to be inspected only if the period of time in which they become accessible is greater than 48 hours.

b. <u>Visual Inspection Acceptance Criteria</u>

Visual inspections shall verify (1) that there are no visible indications of damage or impaired operability, and (2) that the snubber installation exhibits no visual indications of detachment from foundations or supporting structures. Snubbers which appear inoperable as a result of visual inspections may be determined operable for the purpose of establishing the next visual inspection interval, providing that (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers that may be generically susceptible; and (2) the affected snubber is functionally tested in the as-found condition and determined operable per Specification 4.6.I.c, as applicable.

3.6 LIMITING CONDITIONS FOR OPERATION

4.6 SURVEILLANCE REQUIREMENTS

When the fluid port of a hydraulic snubber is found to be uncovered, the snubber shall be determined inoperable unless it can be determined operable via functional testing for the purpose of establishing the next visual inspection interval. The functional test, in this case, shall be started with the piston in the as-found condition, extending the piston rod in the tension mode direction.

c. Functional Tests

At least once per 18 months during shutdown, a representative sample of 10% of the snubbers in use in the plant shall be functionally tested either in place or in a bench test. For each snubber that does not meet the functional test acceptance criteria of Specification 4.6.I.1.d, an additional 10% of the snubbers shall be functionally tested until no more failures are found or until all snubbers have been functionally tested.

3.6 LIMITING CONDITIONS FOR OPERATION

4.6 SURVEILLANCE REQUIREMENTS

Snubbers of a rated capacity greater than the capability of the testing machine shall be functionally tested as follows: (1) the lock up and bleed velocity of the snubber valve shall be verified by testing it on a cylinder that is within the capability of the testing machine, (2) the free stroke of the cylinder shall be checked, and (3) the pressure retaining capability of the cylinder shall be checked.

Snubbers identified as especially difficult to remove or in high radiation areas shall also be included in the representative sample.

In addition to the regular sample, snubbers which failed the previous functional test shall be retested during the next test period unless the root cause for the problem has been determined and corrective actions implemented. If a spare snubber has been installed in place of a failed snubber, then both the failed snubber (if it is repaired and installed in another position) and the spare snubber shall be retested during the next test period.

3.6 LIMITING CONDITIONS FOR OPERATION	4.6 SURVEILLANCE REQUIREMENTS
	Failure of these snubbers shall not entail functional testing of additional snubbers. If any snubber selected for functional testing either fails to lock up or fails to move, i.e., frozen in place, the cause will be evaluated and if caused by manufacturer or design deficiency, all generically susceptible snubbers of the same design subject to the same defect shall be functionally tested. This testing requirement shall be independent of the requirements stated above for snubbers not meeting the functional test acceptance criteria. For the snubber(s) found inoperable, a documented engineering evaluation shall be performed on the component(s) which are supported by the snubber(s). The scope of the evaluation shall be based on engineering judgement and may be limited to a visual inspection of the supported component(s). The purpose of this engineering evaluation shall be to determine if the component(s) supported by the

3.6 LIMITING CONDITIONS FOR OPERATION	4.6 SURVEILLANCE REQUIREMENTS

snubber(s) were adversely affected by the inoperability of the snubber(s) in order to ensure that the supported component remains capable of meeting the designed service.

d. <u>Hydraulic Snubbers Functional Test Acceptance Criteria</u>

The hydraulic snubber functional test shall verify that:

1. Activation (restraining action) is achieved within the specified range of velocity or acceleration in both tension and compression.

2. Snubber bleed, or release rate, where required, is within the specified range in compression or tension. For snubbers specifically required to not displace under continuous load, the ability of the snubber to withstand load without displacement shall be verified.

3.6 LIMITING CONDITIONS FOR OPERATION	4.6 SURVEILLANCE REQUIREMENTS
J. Thermal Hydraulic Stability	J. Thermal Hydraulic Stability
1. When the reactor mode switch is in RUN, the reactor shall not intentionally be operated in a natural circulation mode, except as permitted by Specification 3.6.H.3, nor shall an idle recirculation pump be started with the reactor in a natural circulation mode.	

FIGURE 3.6.1

REACTOR VESSEL PRESSURE-TEMPERATURE LIMITATIONS

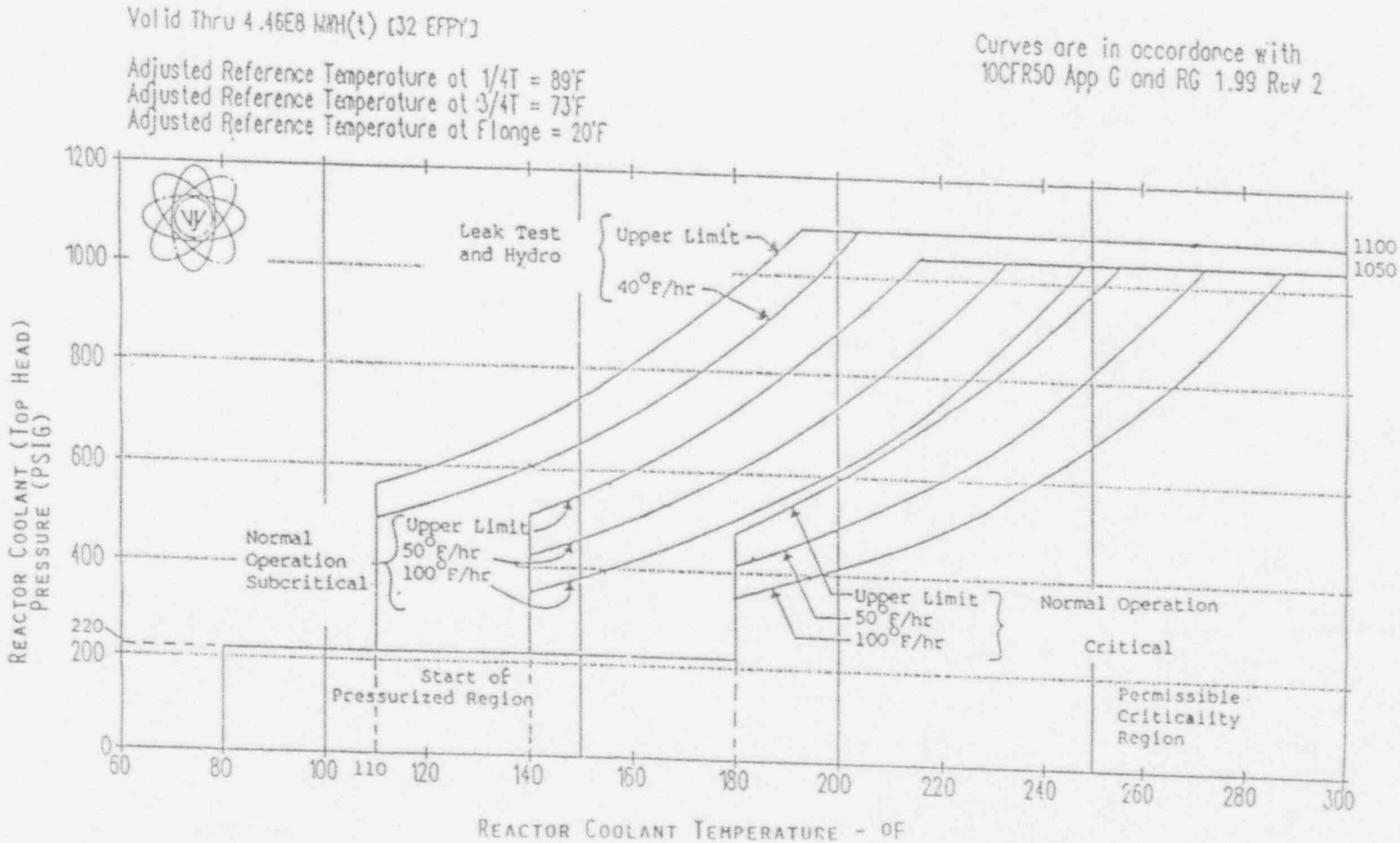
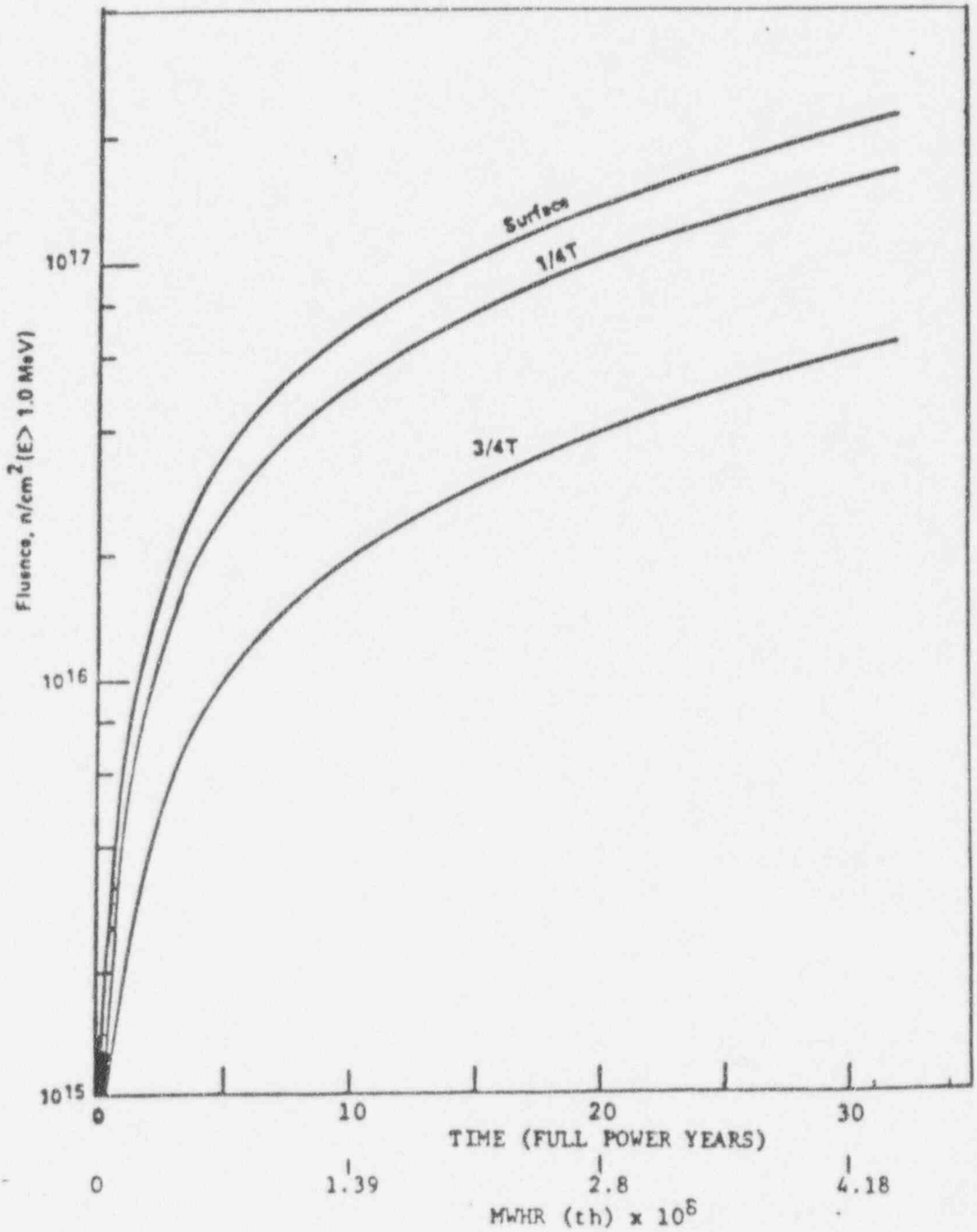


FIGURE 3.6.2

FAST NEUTRON FLUENCE (E>1 MEV) AS A FUNCTION OF THERMAL ENERGY AND FULL POWER YEARS



REFERENCE: L. M. Lowry et al. "Examination, Testing, and Evaluation of Irradiated Pressure Vessel Surveillance Specimens from Vermont Yankee Nuclear Power Station.

Batelle Columbus Laboratories Report #BCL-585-84-3, May 15, 1984

FIGURE 3.6.3

FLUENCE FACTOR FOR USE IN REGULATORY GUIDE 1.99

REVISION 2

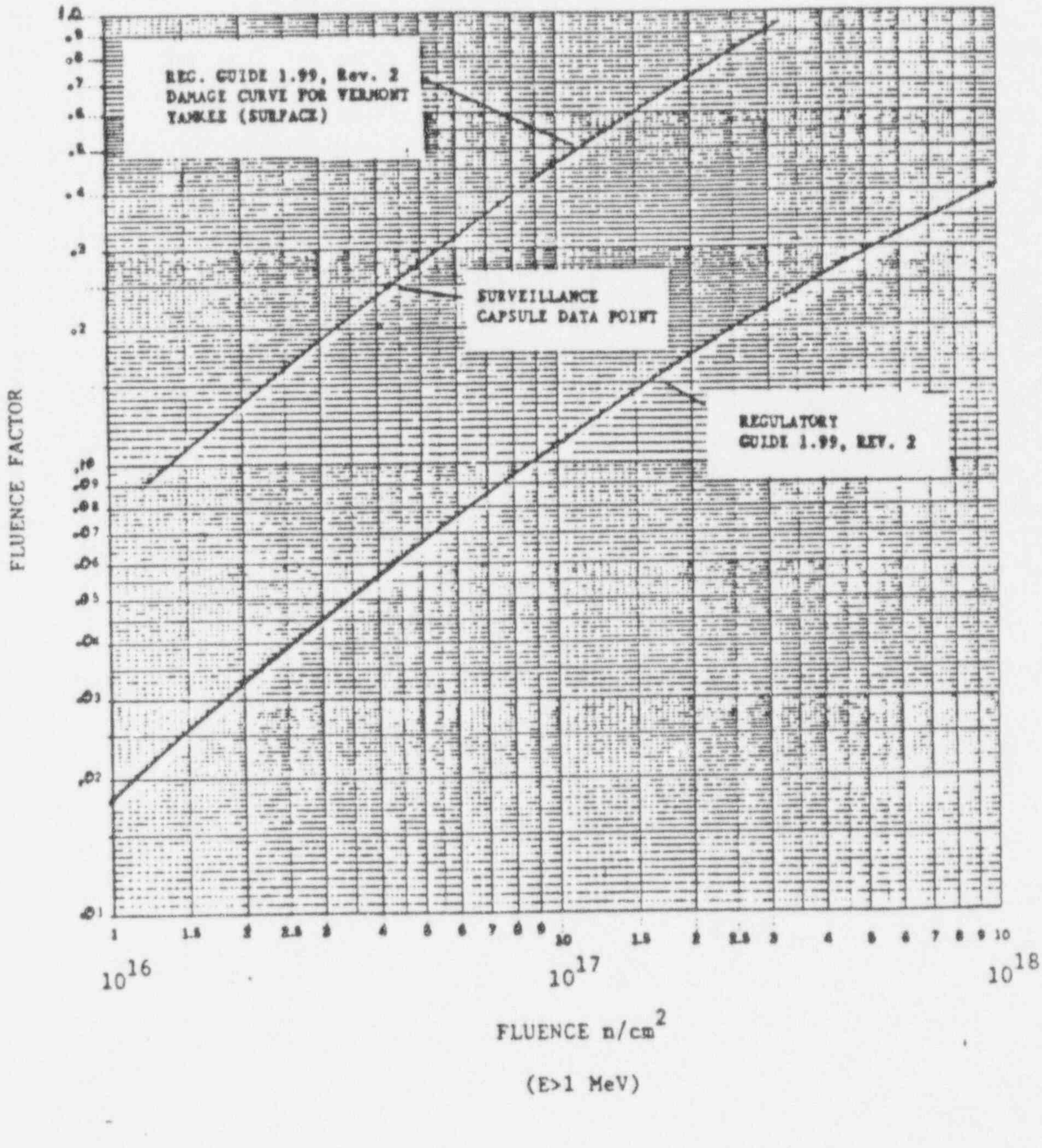
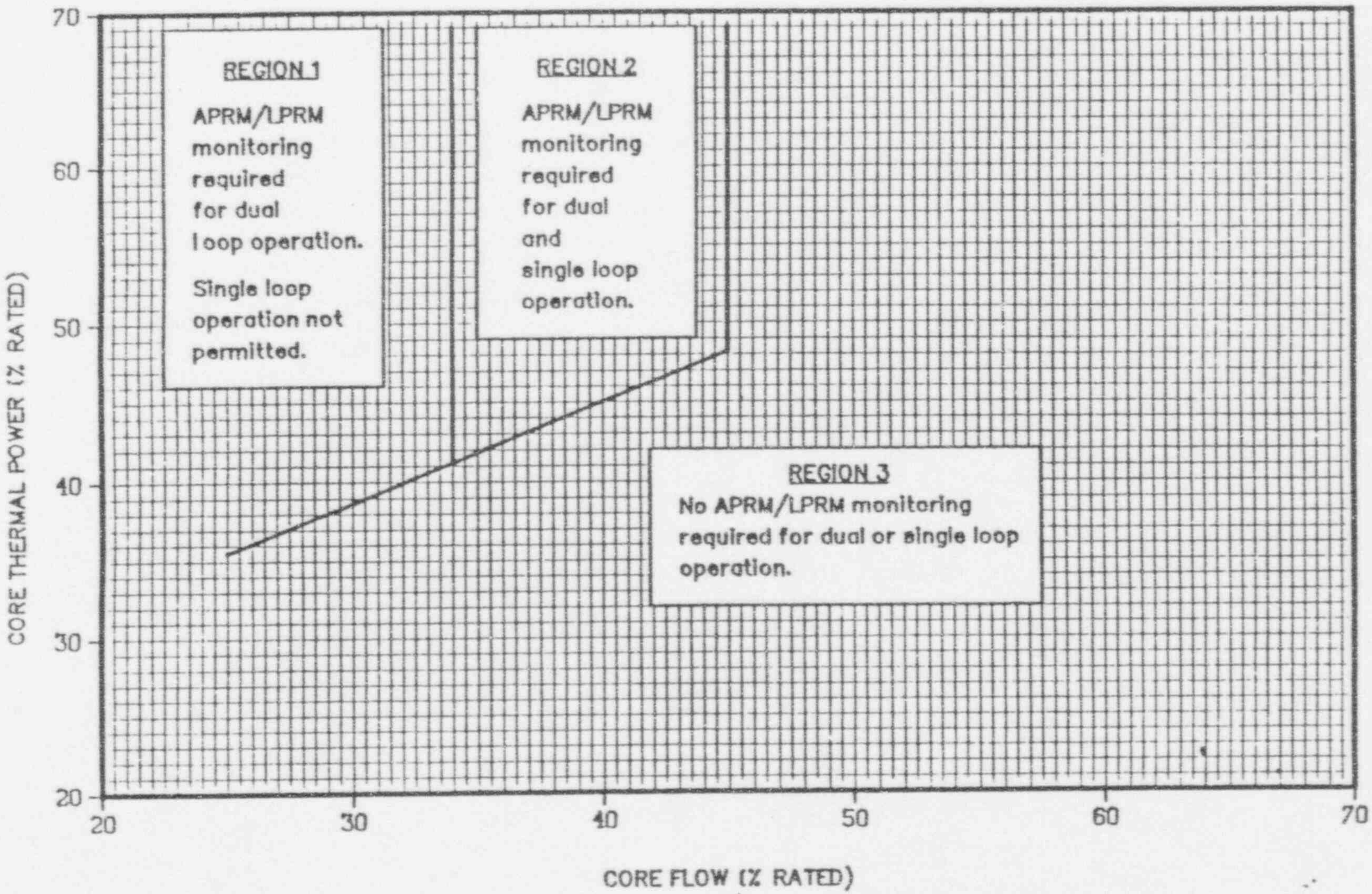


FIGURE 3.6.4

THERMAL POWER AND CORE FLOW LIMITS FOR APRM/LPRM MONITORING



BASES:

3.6 and 4.6 REACTOR COOLANT SYSTEM

A. Pressure and Temperature Limitations

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 4.2 of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

During heatup, the thermal gradients in the reactor vessel wall produce thermal stresses which vary from compressive at the inner wall to tensile at the outer wall. These thermal induced compressive stresses tend to alleviate the tensile stresses induced by their internal pressure. Therefore, a pressure-temperature curve based on steady-state conditions (i.e., no thermal stresses) represents a lower bound of all similar curves for finite heatup rates when the inner wall of the vessel is treated as the governing locations.

The heatup analysis also covers the determination of pressure-temperature limitations for the case in which the outer wall of the vessel becomes the controlling location. The thermal gradients established during heatup produce tensile stresses at the outer wall of the vessel. These stresses are additive to the pressure induced tensile stresses which are already present. The thermal induced stresses at the outer wall of the vessel are tensile and are dependent on both the rate of heatup and the time along the heatup ramp; therefore, a lower bound curve similar to that described for the heatup of the inner wall cannot be defined. Subsequently, for the cases in which the outer wall of the vessel becomes the stress controlling location, each heatup rate of interest must be analyzed on an individual basis.

In order to prevent undue stress on the vessel nozzles and bottom head region, the recirculation loop temperatures should be within 50°F of each other prior to startup of an idle loop.

The reactor vessel materials have been tested to determine their initial reference temperature nil-ductility transition temperature (RT_{NDT}) of 40°F maximum. Reactor operation and resultant fast neutron (E > 1 Mev) irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature can be predicted using current industry practices and Vermont Yankee Surveillance Program data. (Regulatory Guide 1.99, Revision 2, and Battelle Columbus Laboratory Report BCL 585-84-3, dated May 15, 1984. The pressure/temperature limit curve, Figure 3.6.1, includes predicted adjustments for this shift in RT_{NDT} for operation through 4.46×10^8 MWH(t), as well as adjustments for possible errors in the pressure and temperature sensing instruments.

The reference temperature of the closure flange material was determined by material testing and Branch Technical Position MTEB 5-2, "Fracture Toughness Requirements for Older Plants". The closure flange is located in a low neutron fluence area and therefore no measurable RT_{NDT} shift is expected over the life of the plant.

BASES: 3.6 and 4.6 (Cont'd)

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185 reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the irradiation samples and vessel inside radius are essentially identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of the reactor vessel. Battelle Columbus Laboratory Report BCL-585-84-3, dated May 15, 1984, provides this information for the ten-year surveillance capsule. In order to estimate the material properties at the 1/4 and 3/4 T positions in the vessel plate, the shift in RT_{NDT} is determined in accordance with Regulatory Guide 1.99, Revision 2. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule is different from the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

The pressure-temperature limit lines, shown on Figure 3.6.1, for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10CFR50 for reactor criticality and for inservice leak and hydrostatic testing.

The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided to assure compliance with the requirements of Appendix H to CFR Part 50.

Coolant Chemistry

A steady-state radioiodine concentration limit of 1.1 µCi of I-131 dose equivalent per gram of water in the Reactor Coolant System can be reached if the gross radioactivity in the gaseous effluents is near the limit, as set forth in Specification 3.8.C.1a, or there is a failure or prolonged shutdown of the cleanup demineralizer. In the event of a steam line rupture outside the drywell, the NRC staff calculations show the resultant radiological dose at the site boundary to be less than 30 Rem to the thyroid. This dose was calculated on the basis of the radioiodine concentration limit of 1.1 µCi of I-131 dose equivalent per gram of water, atmospheric diffusion from an equivalent elevated release of 10 meters at the nearest site boundary (190 m) for a $X/Q = 3.9 \times 10^{-3}$ sec/m^3 (Pasquill D and 0.33 m/sec equivalent), and a steam line isolation valve closure time of five seconds with a steam/water mass release of 30,000 pounds.

The iodine spike limit of four (4) microcuries of I-131 dose equivalent per gram of water provides an iodine peak or spike limit for the reactor coolant concentration to assure that the radiological consequences of a postulated LOCA are within 10CFR Part 100 dose guidelines.

The reactor coolant sample will be used to assure that the limit of Specification 3.6.B.1 is not exceeded. The radioiodine concentration would not be expected to change rapidly during steady-state operation over a period of 96 hours. In addition, the trend of the radioactive gaseous effluents, which is continuously monitored, is a good indicator of the trend of the radioiodine concentration in the reactor coolant. When a significant increase in radioactive gaseous effluents is indicated, as specified, an additional reactor coolant sample shall be taken and analyzed for radioactive iodine.

BASES: 3.6 and 4.6 (Cont'd)

Whenever an isotopic analysis is performed, a reasonable effort will be made to determine a significant percentage of those contributors representing the total radioactivity in the reactor coolant sample. Usually at least 80 percent of the total gamma radioactivity can be identified by the isotopic analysis.

It has been observed that radioiodine concentration can change rapidly in the reactor coolant during transient reactor operations, such as reactor shutdown, reactor power changes, and reactor startup if failed fuel is present. As specified, additional reactor coolant samples shall be taken and analyzed for reactor operations in which steady-state radioiodine concentrations in the reactor coolant indicate various levels of iodine releases from the fuel. Since the radioiodine concentration in the reactor coolant is not continuously measured, reactor coolant sampling would be ineffective as a means to rapidly detect gross fuel element failures. However, some capability to detect gross fuel element failures is inherent in the radiation monitors in the off-gas system on the main steam line.

Materials in the primary system are primarily 304 stainless steel and Zircaloy. The reactor water chemistry limits are established to prevent damage to these materials. The limit placed on chloride concentration is to prevent stress corrosion cracking of the stainless steel.

When conductivity is in its proper normal range (approximately 10 µmho/cm during reactor startup and 5 µmho/cm during power operation), pH and chloride and other impurities affecting conductivity must also be within their normal range. When and if conductivity becomes abnormal, then chloride measurements are made to determine whether or not they are also out of their normal operating values. This would not necessarily be the case. Conductivity could be high due to the presence of a neutral salt, e.g., Na_2SO_4, which would not have an effect on pH or chloride. In such a case, high conductivity alone is not a cause for shutdown. In some types of water-cooled reactors, conductivities are in fact high due to purposeful addition of additives. In the case of BWRs, however, no additives are used and where neutral pH is maintained, conductivity provides a very good measure of the quality of the reactor water. Significant changes therein provide the operator with a warning mechanism so he can investigate and remedy the condition causing the change before limiting conditions, with respect to variables affecting the boundaries of the reactor coolant, are exceeded. Methods available to the operator for correcting the off-standard condition include operation of the reactor cleanup system reducing the input of impurities and placing the reactor in the cold shutdown condition. The major benefit of cold shutdown is to reduce the temperature dependent corrosion rates and provide time for the cleanup system to reestablish the purity of the reactor coolant. During startup periods, which are in the category of less than 100,000 pounds per hour, conductivity may exceed 5 µmho/cm because of the initial evolution of gases and the initial addition of dissolved metals. During this period of time when the conductivity exceed 5 µmho (other than short term spikes), samples will be taken to assure the chloride concentration is less than 0.1 ppm.

The conductivity of the reactor coolant is continuously monitored. The samples of the coolant which are taken every 96 hours will serve as a reference for calibration of these monitors and is considered adequate to assure accurate readings of the monitors. If conductivity is within its normal range, chlorides and other

BASES: 3.6 and 4.6 (Cont'd)

impurities will also be within their normal ranges. The reactor cooling samples will also be used to determine the chlorides. Therefore, the sampling frequency is considered adequate to detect long-term changes in the chloride ion content. Isotopic analyses required by Specification 4.6.B.2 may be performed by a gamma scan and gross beta and alpha determination.

The conductivity of the feedwater is continuously monitored and alarm set points consistent with Regulatory requirements given in Regulatory Guide 1.56, "Maintenance of Water Purity in Boiling Water Reactors," have been determined. The results from the conductivity monitors on the feedwater can be correlated with the results from the conductivity monitors on the reactor coolant water to indicate demineralizer breakthrough and subsequent conductivity levels in the reactor vessel water.

C. Coolant Leakage

The 5 gpm limit for unidentified leaks was established assuming such leakage was coming from the reactor coolant system. Tests have been conducted which demonstrate that a relationship exists between the size of a crack and the probability that the crack will propagate. These tests suggest that for leakage somewhat greater than the limit specified for unidentified leakage; the probability is small that imperfections or cracks associated with such leakage would grow rapidly. Leakage less than the limit specified can be detected within a few hours utilizing the available leakage detection systems. If the limit is exceeded and the origin cannot be determined in a reasonably short time the plant should be shutdown to allow further investigation and corrective action.

The removal capacity from the drywell floor drain sump and the equivalent drain sump is 50 gpm each. Removal of 50 gpm from either of these sumps can be accomplished with considerable margin.

D. Safety and Relief Valves

Parametric evaluations have shown that only three of the four relief valves are required to provide a pressure margin greater than the recommended 25 psi below the safety valve actuation settings as well as a MCPR > 1.06 for the limiting overpressure transient below 98% power. Consequently, 95% power has been selected as a limiting power level for three valve operation. For the purposes of this limiting condition a relief valve that is unable to actuate within tolerance of its set pressure is considered to be as inoperable as a mechanically malfunctioning valve.

Experience in safety valve operation shows that a testing of 50% of the safety valves per refueling outage is adequate to detect failures or deterioration. The tolerance value is specified in Section III of the ASME Boiler and Pressure Vessel Code as ±1% of design pressure. An analysis has been performed which shows that with all safety valves set 1% higher the reactor coolant pressure safety limit of 1375 psig is not exceeded.

E. Structural Integrity and Operability Testing

A pre-service inspection of the components listed in Table 4.2-4 of the FSAR was conducted after site erection to assure freedom from defects greater than code allowance; in addition, this serves as a reference base for further inspections. Prior to operation, the reactor primary system was free of gross defects. In addition, the facility has been designed such that gross defects should not occur

BASES: 3.6 and 4.6 (Cont'd)

throughout plant life. The inservice inspection and testing programs are performed in accordance with 10CFR50, Section 50.55a(g) except where specific relief has been granted by the NRC. These inspection and testing programs provide further assurance that gross defects are not occurring and ensure that safety-related components remain operable.

The type of inspection planned for each component depends on location, accessibility, and type of expected defect. Direct visual examination is proposed wherever possible since it is sensitive, fast, and reliable. Magnetic particle and liquid penetrant inspections are planned where practical, and where added sensitivity is required. Ultrasonic testing and radiography shall be used where defects can occur on concealed surfaces.

The in-service inspection and testing programs presented at this time are based on a thorough evaluation of present technology and state-of-the-art inspection and testing techniques.

F. Jet Pumps

Failure of a jet pump nozzle assembly hold down mechanism, nozzle assembly and/or riser, would increase the cross-sectional flow area for blowdown following the design bases double-ended line break. Therefore, if a failure occurred, repairs must be made.

The detection technique is as follows. With the two recirculation pumps balanced in speed to within ±5%, the flow rates in both recirculation loops will be verified by main Control Room monitoring instruments. If the two flow rate values do not differ by more than 10%, riser and nozzle assembly integrity has been verified. If they do differ by 10% or more the core flow rate measured by the jet pump diffuser differential pressure system must be checked against the core flow rate derived from the measured value of loop flow to core flow correlation. If the difference between measured and derived core flow rate is 10% or more (with the measured value higher) diffuser measurements will be taken to define the location within the vessel of failed jet pump nozzle (or riser) and the plant shut down for repairs. If the potential blowdown flow area is increased, the system resistance to the recirculation pump is also reduced; hence, the affected drive pump will "run out" to a substantially higher flow rate (approximately 115% to 120% for a single nozzle failure). If the two loops are balanced in flow at the same pump speed, the resistance characteristics cannot have changed. Any imbalance between drive loop flow rates would be indicated by the plant process instrumentation. In addition, the affected jet pump would provide a leakage path past the core thus reducing the core flow rate. The reverse flow through the inactive jet pump would still be indicated by a positive differential pressure but the net effect would be a slight decrease (3% to 6%) in the total core flow measure. This decrease, together with the loop flow increase, would result in a leak of correlation between measured and derived core flow rate.

The following factors form the basis for the surveillance requirements:

- A break in a jet pump decreases the flow resistance characteristic of the external piping loop causing the recirculation pump to operate at a higher flow condition when compared to previous operation.

BASES: 3.6 and 4.6 (Cont'd)

- The change in flow rate of the failed jet pump produces a change in the indicated flow rate of that pump relative to the other pumps in that loop. Comparison of the data with a normal relationship or pattern provides the indication necessary to detect a failed jet pump.

- The jet pump flow deviation pattern derived from the diffuser to lower plenum differential pressure readings will be used to further evaluate jet pump operability in the event that the jet pumps fail the tests in Specifications 4.6.F.1 and 2.

Agreement of indicated core flow with established power-core flow relationships provides the most assurance that recirculation flow is not bypassing the core through inactive or broken jet pumps. This bypass flow is reverse with respect to normal jet pump flow. The indicated total core flow is a summation of the flow indications for the twenty individual jet pumps. The total core flow measuring instrumentation sums reverse jet pump flow as though it were forward flow. Thus, the indicated flow is higher than actual core flow by at least twice the normal flow through any backflowing pump. Reactivity inventory is known to a high degree of confidence so that even if a jet pump failure occurred during a shutdown period, subsequent power ascension would promptly demonstrate abnormal control rod withdrawal for any power-flow operating map point.

A nozzle-riser system failure could also generate the coincident failure of a jet pump body; however, the converse is not true. The lack of any substantial stress in the jet pump body makes failure impossible without an initial nozzle-riser system failure.

G. Single Loop Operation

Continuous operation with one recirculation loop was justified in "Vermont Yankee Nuclear Power Station Single Loop Operation", NEDO-30060, February 1983, with the adjustments specified in Technical Specification 3.6.G.1.a.

APRM and/or LPRM oscillations in excess of those specified in Section 3.6.G.1.b could be an indication that a condition of thermal hydraulic/neutronic instability exists and that appropriate remedial action should be taken. By restricting core flow to greater than or equal to 34% of rated, which corresponds to the core flow at the 80% rod line with 2 recirculation pumps running at minimum speed, the region of the power/flow map where these oscillations are most likely to occur is avoided (Region 1 of Figure 3.6.4). These specifications are based upon the guidance of GE SIL #380, Revision 1, dated February 10, 1984.

During single loop operation, the idle recirculation loop is isolated by electrically disarming the recirculation pump motor generator set drive motor, until ready to resume two loop operation. This is done to prevent a cold water injection transient caused by an inadvertent pump startup.

Under single loop operation, the flow control is placed in the manual mode to avoid control oscillations which may occur in the recirculation flow control system under these conditions.

H. Recirculation System

The largest recirculation break area assumed in the ECCS evaluation was 4.14 square feet.

BASES: 3.6 and 4.6 (Cont'd)

APRM and/or LPRM oscillations in excess of those specified in Section 3.6.H.1 could be an indication that a condition of thermal hydraulic instability exists and that appropriate remedial action should be taken. These specifications are based upon the guidance of GE SIL #380, Revision 1, dated February 10, 1984.

Specification 3.6.H.3 restricts reactor operation under natural circulation conditions in order to avoid potential thermal hydraulic/neutronic instabilities.

I. Shock Suppressors (Snubbers)

All snubbers are required operable to ensure that the structural integrity of the Reactor Coolant System and all other safety-related systems is maintained during and following a seismic or other event initiating dynamic loads.

The visual inspection frequency is based upon maintaining a constant level of snubber protection to systems. Therefore, the required inspection interval varies inversely with the observed snubber failures and is determined by the number of inoperable snubbers found during an inspection. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

When the cause of the rejection of a snubber is clearly established and remedied for that snubber and for any other snubbers that may be generically susceptible, and verified by functional testing, that snubber may be exempted from being counted as inoperable. Generically susceptible snubbers are those which are (1) of a specific make or model, (2) of the same design, and (3) similarly located or exposed to the same environmental conditions such as temperature, radiation, and vibration. These characteristics of the snubber installation shall be evaluated to determine if further functional testing of similar snubber installations is warranted.

When a snubber is found inoperable, an engineering evaluation is performed, in addition to the determination of the snubber mode of failure, in order to determine if any safety-related component or system has been adversely affected by the inoperability of the snubber. The engineering evaluation shall determine whether or not the snubber mode of failure has imparted a significant effect or degradation on the supported component or system.

To provide assurance of snubber functional reliability, a representative sample of the installed snubbers will be functionally tested once each operating cycle. Observed failures of these sample snubbers shall require functional testing of additional units.

J. Thermal Hydraulic Stability

Not allowing startup of an idle recirculation pump from natural circulation conditions prevents the reactivity insertion transient that would occur.

3.7 LIMITING CONDITIONS FOR OPERATION

3.7 STATION CONTAINMENT SYSTEMS

Applicability:

Applies to the operating status of the primary and secondary containment systems.

Objective:

To assure the integrity of the primary and secondary containment systems.

Specification:

A. Primary Containment

1. Whenever primary containment is required, the volume and temperature of the water in the suppression chamber shall be maintained within the following limits:

    a. Maximum Water Temperature during normal operation - 100°F.

    b. Maximum Water Temperature during any test operation which adds heat to the suppression pool - 100°F.

    c. If Torus Water Temperature exceeds 110°F, initiate an immediate scram of the reactor. Power operation shall not be resumed until the pool temperature is reduced below 100°F.

    d. During reactor isolation conditions, the reactor pressure vessel shall be depressurized to less than 200 psig at normal cooldown rates if the torus water temperature exceeds 120°F.

4.7 SURVEILLANCE REQUIREMENTS

4.7 STATION CONTAINMENT SYSTEMS

Applicability:

Applies to the primary and secondary containment system integrity.

Objective:

To verify the integrity of the primary and secondary containments.

Specification:

A. Primary Containment

1. The suppression chamber water level and temperature shall be checked once per shift. A visual inspection of the suppression chamber interior including water line regions and the interior painted surfaces above the water line shall be made at each refueling outage. Whenever there is indication of relief valve operation which adds heat to the suppression pool, the pool temperature shall be continually monitored and also observed and logged every 5 minutes until the heat addition is terminated. Whenever there is indication of relief valve operation with the temperature of the suppression pool reaching 160°F or more and the primary coolant system pressure greater than 200 psig, an external visual examination of the suppression chamber shall be conducted before resuming power operation.

3.7 LIMITING CONDITIONS FOR OPERATION

e. Minimum Water Volume - 68,000 cubic feet

f. Maximum Water Volume - 70,000 cubic feet

2. Primary containment integrity shall be maintained at all times when the reactor is critical or when the reactor water temperature is above 212°F and fuel is in the reactor vessel except while performing low power physics tests at atmospheric pressure at power levels not to exceed 5 Mw(t).

3. Whenever primary containment is required, the total primary containment leakage rate shall not exceed 0.8 weight percent per day (L_a) at a pressure of 44 psig (P_a).

4. Whenever primary containment is required, the leakage from any one isolation valve shall not exceed 5 percent of the maximum allowable leak rate (L_a) at peak accident pressure (P_a) and the leakage from any one main steam line isolation valve shall not exceed 15.5 scf/hr at 44 psig (P_a).

4.7 SURVEILLANCE REQUIREMENTS

2. The primary containment integrity shall be demonstrated as required by Appendix J to 10 CFR Part 50. The primary containment shall meet the containment acceptance requirements set forth in that appendix.

   a. Penetrations and seals listed in Table 4.7.1 shall be leak tested at 44 psig (Pa).

   b. Type C tests shall be performed on the isolation valves listed in Table 4.7.2.a.

3. Prior to violating the integrity of a system outside the primary containment, which is connected to any valve listed in Table 4.7.2b, the isolation valves bounding the opening shall have Type C tests performed. If the opening cannot be isolated from the containment by two isolation valves which meet the acceptance criteria of Appendix J (10CFR Part 50), a blank flange shall be installed on the opening.

4. The leakage from any one isolation valve shall not exceed 5% of Ltm. The leakage from any one main steam line isolation valve shall not exceed 11.5 scf/hr at 24 psig (Pt). Repair and retest shall be conducted to insure compliance.

3.7 LIMITING CONDITIONS FOR OPERATION	4.7 SURVEILLANCE REQUIREMENTS

3.7 LIMITING CONDITIONS FOR OPERATION

5. Pressure Suppression Chamber-Reactor Building Vacuum Breakers

   a. Two of two pressure suppression chamber-reactor building vacuum breaker systems shall be operable at all times when the primary containment integrity is required. The setpoint of the differential pressure instrumentation which actuates the pressure suppression chamber-reactor building air-operated vacuum breakers shall be $\leq$0.5 psid. The self actuating vacuum breakers shall open fully when subjected to a force equivalent to or less than 0.5 psid acting on the valve disk.

   b. With one Reactor Building - Suppression chamber vacuum breaker inoperable for opening but known to be closed, restore the inoperable vacuum breaker to OPERABLE status within seven (7) days or then be in cold shutdown within the following 24 hours.

   c. With on Reactor Building - Suppression chamber vacuum breaker failed open - power operation may continue provided the other vacuum breaker in that

4.7 SURVEILLANCE REQUIREMENTS

5. Pressure Suppression Chamber - Reactor Building Vacuum Breakers

   a. The pressure suppression chamber-Reactor Building vacuum breaker instrumentation including setpoint shall be checked for proper operation every three months.

   b. Operability testing of the vacuum breakers shall be in accordance with Specification 4.6.E. Each vacuum breaker shall be tested to determine that the force required to open the vacuum breaker does not exceed the force specified in Specification 3.7.A.5.a and each vacuum breaker shall be inspected and verified to meet design requirements.

3.7 LIMITING CONDITIONS FOR OPERATION

line is verified to be closed and conditions required by 3.7.D.2 are met.

6. Pressure Suppression Chamber - Drywell Vacuum Breakers

   a. When primary containment is required, all suppression chamber - drywell vacuum breakers shall be operable except during testing and as stated in Specifications 3.7.A.6.b and c, below. Suppression chamber - drywell vacuum breakers shall be considered operable if:

      (1) The valve is demonstrated to open fully with the applied force at all valve positions not exceeding that equivalent to 0.5 psi acting on the suppression chamber face of the valve disk.

      (2) The valve can be closed by gravity, when released after being opened by remote or manual means, to within not greater than the equivalent of 0.05 inch at all points along the seal surface of the disk.

4.7 SURVEILLANCE REQUIREMENTS

6. Pressure Suppression Chamber - Drywell Vacuum Breakers

   a. Periodic Operability Tests

      Operability testing of the vacuum breakers shall be in accordance with Specification 4.6.E and following any release of energy to the suppression chamber. Operability of the corresponding position switches and position indicators and alarms shall be verified monthly and following any maintenance.

   b. Refueling Outage Tests

      (1) All suppression chamber - drywell vacuum breaker position indication and alarm systems shall be calibrated and functionally tested.

      (2) At least two (2) of the suppression chamber - drywell vacuum breakers shall be inspected.

3.7 LIMITING CONDITIONS FOR OPERATION

(3) The position alarm system will annunciate in the control room if the valve opening exceeds the equivalent of 0.05 inch at all points along the seal surface of the disk.

b. Up to two (2) of the ten (10) suppression chamber - drywell vacuum breakers may be determined to be inoperable provided that they are secured, or known to be, in the closed position.

c. Reactor operation may continue for fifteen (15) days provided that at least one position alarm circuit for each vacuum breaker is operable and each suppression chamber - drywell vacuum breaker is physically verified to be closed immediately and daily thereafter.

7. <u>Oxygen Concentration</u>

a. The primary containment atmosphere shall be reduced to less than 4 percent oxygen with nitrogen gas during reactor power operation with reactor coolant pressure above 90 psig, except as specified in Specification 3.7.A.7.b.

4.7 SURVEILLANCE REQUIREMENTS

If deficiencies are found such that Specification 3.7.A.6 could not be met, all vacuum breakers shall be inspected and deficiencies corrected.

(3) A drywell to suppression chamber leak rate test shall demonstrate that with an initial differential pressure of not less than 1.0 psi, the differential pressure decay rate shall not exceed the equivalent of the leakage rate through a 1-inch orifice.

7. <u>Oxygen Concentration</u>

The primary containment oxygen concentration shall be measured and recorded on a weekly basis.

3.7 LIMITING CONDITIONS FOR OPERATION

b. Within the 24-hour period subsequent to placing the reactor in the Run mode following a shutdown, the containment atmosphere oxygen concentration shall be reduced to less than 4 percent and maintained in this condition. Deinerting may commence 24 hours prior to a shutdown.

8. If Specification 3.7.A.1 through 3.7.A.7 cannot be met, an orderly shutdown shall be initiated immediately and the reactor shall be in a cold shutdown condition within 24 hours.

9. Drywell/Suppression Chamber d/p

a. Differential pressure between the drywell and suppression chamber shall be maintained $\geq$1.7 psi except as specified in 3.7.A.9.b and 3.7.A.9.c below.

b. The $\geq$1.7 psi differential pressure shall be established within 24 hours of achieving operating pressure and temperature. The differential pressure may be reduced to <1.7 psi 24 hours prior to commencing a cold shutdown.

c. The differential pressure may be reduced to <1.7 psi for a maximum of four hours (period to begin when the

4.7 SURVEILLANCE REQUIREMENTS

9. Drywell/Suppression Chamber d/p

a. The differential pressure between the drywell and suppression chamber shall be recorded once per shift.

b. The operability of the low differential pressure alarm shall be verified once per week.

3.7 LIMITING CONDITIONS FOR OPERATION

ΔP is reduced to <1.7) during required operability testing of the HPCI system pump, the RCIC system pump, the drywell-suppression chamber vacuum breakers, and the suppression chamber-reactor building vacuum breakers, and SBGTS testing.

d. If the specifications of 3.7.A.9.a cannot be met, and the differential pressure cannot be restored within the subsequent six (6) hour period, an orderly shutdown shall be initiated and the reactor shall be in a Hot Shutdown condition in six (6) hours and a Cold Shutdown condition in the following eighteen (18) hours.

B. Standby Gas Treatment System

1. Except as specified in Specification 3.7.B.3 below, both circuits of the standby gas treatment system and the diesel generators required for operation of such circuits shall be operable at all times when secondary containment integrity is required.

4.7 SURVEILLANCE REQUIREMENTS

B. Standby Gas Treatment System

1. At least once per operating cycle, not to exceed 18 months, the following conditions shall be demonstrated.

   a. Pressure drop across the combined HEPA and charcoal filter banks is less than 6 inches of water at 1500 cfm ±10%.

   b. Inlet heater input is at least 9 kW.

3.7 LIMITING CONDITIONS FOR OPERATION

2. a. The results of the in-place cold DOP and halogenated hydrocarbon tests at design flows on HEPA and charcoal filter banks shall show $\geq$99% DOP removal and $\geq$99% halogenated hydrocarbon removal.

b. The results of laboratory carbon sample analysis shall show $\geq$95% radioactive methyl iodide removal (130°C, 95% RH)

c. System fans shall be shown to operate within ±10% of design flow.

4.7 SURVEILLANCE REQUIREMENTS

2. a. The tests and sample analysis of Specification 3.7.B.2 shall be performed initially and at least once per operating cycle not to exceed 18 months, and following painting, fire or chemical release in any ventilation zone communicating with the system, while the system is operating, that could contaminate the HEPA filters or charcoal absorbers.

b. Cold DOP testing shall be performed after each complete or partial replacement of the HEPA filter bank.

c. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of the charcoal filter bank.

In addition, the sample analysis of Specification 3.7.B.2.b and the halogenated hydrocarbon test shall be performed after every 720 hours of system operation.

d. Each circuit shall be operated with the heaters on at least 10 hours every month.

e. An ultrasonic leak test shall be performed on the gaskets sealing the housing panels downstream of the HEPA filters and adsorbers at least

3.7 LIMITING CONDITIONS FOR OPERATION

3. From and after the date that one circuit of the standby gas treatment system is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days unless such circuit is sooner made operable, provided that during such seven days all active components of the other standby gas treatment circuit shall be operable.

4. If this condition cannot be met, procedures shall be initiated immediately to establish the conditions listed in Specifications 3.7.C.1(a) through (d), and compliance shall be completed within 24 hours thereafter.

4.7 SURVEILLANCE REQUIREMENTS

once per operating cycle not to exceed 18 months. If the ultrasonic test indicates the presence of a leak, the condition will be evaluated and the gasket repaired or replaced as necessary.

f. DOP and halogenated hydrocarbon test shall be performed following any design modification to the Standby Gas Treatment System housing that could have an effect on the filter efficiency.

g. An air distribution test demonstrating uniformity within ±20% across the HEPA filters and charcoal adsorbers shall be performed if the SBGTS housing is modified such that air distribution could be affected.

3. a. At least once per operating cycle automatic initiation of each branch of the Standby Gas Treatment System shall be demonstrated.

b. Operability testing of valves shall be in accordance with Specification 4.6.E.

c. When one circuit of the Standby Gas Treatment System is made or found to be inoperable, the other circuit shall have been or shall be demonstrated to be operable within 24 hours.

3.7 LIMITING CONDITIONS FOR OPERATION

C. Secondary Containment System

1. Integrity of the secondary containment system shall be maintained during all modes of plant operation except when all of the following conditions are met:

   a. The reactor is subcritical and Specification 3.3.A is met and

   b. The reactor water temperature is below 212°F and the reactor coolant system is vented.

   c. No activity is being performed which can reduce the shutdown margin below that specified in Specification 3.3.A.

   d. The fuel cask or irradiated fuel is not being moved in the Reactor Building.

4.7 SURVEILLANCE REQUIREMENTS

C. Secondary Containment System

1. Surveillance of secondary containment shall be performed as follows:

   a. A preoperational secondary containment capability test shall be conducted after isolating the Reactor Building and placing either Standby Gas Treatment System filter train in operation. Such tests shall demonstrate the capability to maintain a 0.15 inch of water vacuum under calm wind ($2 < u < 5$ mph) condition with a filter train flow rate of not more than 1500 cfm.

   b. Additional tests shall be performed during the first operating cycle under an adequate number of different environmental wind conditions to enable valid extrapolation of the test results.

   c. Secondary containment capability to maintain a 0.15 inch of water vacuum under calm wind ($2<\bar{u}<5$ mph) conditions with a filter train flow rate of not more than 1,500 cfm, shall be demonstrated at least quarterly and at each refueling outage prior to refueling.

3.7 LIMITING CONDITIONS FOR OPERATION

2. Core spray and LPCI pump lower compartment door openings shall be closed at all times except during passage or when reactor coolant temperature is less than 212°F.

D. Primary Containment Isolation Valves

1. During reactor power operating conditions all isolation valves listed in Table 4.7.2 and all instrument line flow check valves shall be operable except as specified in Specification 3.7.D.2.

4.7 SURVEILLANCE REQUIREMENTS

2. The core spray and LPCI lower compartment openings shall be checked closed daily.

D. Primary Containment Isolation Valves

1. Surveillance of the primary containment isolation valves should be performed as follows:

   a. The operable isolation valves that are power operated and automatically initiated shall be tested for automatic initiation and the closure times specified in Table 4.7.2 at least once per operating cycle.

   b. Operability testing of the primary containment isolation valves shall be performed in accordance with Specification 4.6.E.

   c. At least once per quarter, with the reactor power less than 75 percent of rated, trip all main steam isolation valves (one at a time) and verify closure time.

   d. At least twice per week, the main steam line isolation valves shall be exercised by partial closure and subsequent reopening.

3.7 LIMITING CONDITIONS FOR OPERATION

2. In the event any isolation valve specified in Table 4.7.2 becomes inoperable, reactor power operation may continue provided at least one valve in each line having an inoperable valve is in the mode corresponding to the isolated condition.

3. If Specifications 3.7.D.1 and 3.7.D.2 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours.

4.7 SURVEILLANCE REQUIREMENTS

2. Whenever an isolation valve listed in 4.7.2 is inoperable, the position of at least one other valve in each line having an inoperable valve shall be logged daily.

TABLE 4.7.1

PENETRATIONS AND SEALS SUBJECT TO TYPE B TESTING

Penetration Number	Identification	Number of Penetrations
X-7A, D	Main Steam Line A, D	4
X-9A, B	Feedwater Line A, B	2
X-11	HPCI Steam Line	1
X-12	Shutdown Cooling Supply	1
X-13A, B	RHR Return to Reactor	2
X-14	Supply to Reactor Water Cleanup	1
X-16A, B	Core Spray to Reactor	2
X-1	Equipment Access Hatch	1
X-3	Drywell Head Flange	1
X-4	Drywell Head Access Hatch	2
X-6	CRD Removal Hatch	1
SLH-A, H	Shear Lug Access Covers	8
X-202A, H & J, K	Vacuum Relief Access Covers	10
X-213A, B	Torus Drains	2
X-200A, B	Torus Manways	2

TABLE 4.7.2.a

PRIMARY CONTAINMENT ISOLATION VALVES

VALVES SUBJECT TO TYPE C LEAKAGE TESTS

Isolation Group (1)	Valve Identification	Number of Power Operated Valves Inboard	Number of Power Operated Valves Outboard	Maximum Operating Time (sec)	Normal Position	Action on Initiating Signal
1	Main Steam Line Isolation (2-80A, D & 2-86A, D)	4	4	5 (Note 2)	Open	GC
1	Main Steam Line Drain (2-74, 2-77)	1	1	35	Closed	SC
1	Recirculation Loop Sample Line (2-39, 2-40)	1	1	5	Closed	SC
2	RHR Discharge to Radwaste (10-57, 10-66)		2	25	Closed	SG
2	Drywell Floor Drain (20-82, 20-83)		2	20	Open	GC
2	Drywell Equipment Drain (20-94, 20-95)		2	20	Open	GC
3	Drywell Air Purge Inlet (16-19-9)		1	10	Closed	SC
3	Drywell Air Purge Inlet (16-19-8)		1	10	Open	GC
3	Drywell Purge & Vent Outlet (16-19-7A)		1	10	Closed*	SC
3	Drywell Purge & Vent Outlet Bypass (16-19-6A)		1	10	Closed	SC
3	Drywell & Suppression Chamber Main Exhaust (16-19-7)		1	10	Closed*	SC
3	Suppression Chamber Purge Supply (16-19-10)		1	10	Closed	SC
3	Suppression Chamber Purge & Vent Outlet (16-19-7B)		1	10	Closed	SC
3	Suppression Chamber Purge & Vent Outlet Bypass (16-19-6B)		1	10	Open	GC

* Valves 16-19-7 and 16-19-7A shall have stops installed to limit valve opening to 50° or less.

TABLE 4.7.2.a
(Cont'd)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVES SUBJECT TO TYPE C LEAKAGE TESTS

Isolation Group (1)	Valve Identification	Number of Power Operated Valves Inboard	Outboard	Maximum Operating Time (sec)	Normal Position	Action on Initiating Signal
3	Exhaust to Standby Gas Treatment System (16-19-6)		1	10	Open	GC
3	Containment Purge Supply (16-19-23)		1	10	Open	GC
3	Containment Purge Makeup (16-20-20, 16-20-22A, 16-20-22B)		3	NA	Closed	SC
5	Reactor Cleanup System (12-15 12-18)	1	1	25	Open	GC
6	HPCI (23-15, 23-16)	1	1	55	Open	GC
6	RCIC (13-15 13-16)	1	1	20	Open	GC
	Primary/Secondary Vacuum Relief (16-19-11A, 16-19-11B)		2	NA	Closed	SC
	Primary/Secondary Vacuum Relief (16-19-12A, 16-19-12B)		2	NA	Closed	Process
3	Containment Air Sampling (VG 23, VG 26, 109-76A&B)		4	5	Open	GC
	Feedwater Check Valves (V2-27A, -96A, -28A, -28B)			NA	Open	Process

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TABLE 4.7.2.b

PRIMARY CONTAINMENT ISOLATION VALVES

VALVES NOT SUBJECT TO TYPE C LEAKAGE TESTS

Isolation Group (1)	Valve Identification	Number of Power Operated Valves		Maximum Operating Time (sec)	Normal Position	Action on Initiating Signal
		Inboard	Outboard			
2	RHR Return to Suppression Pool (10-39A, B)		2	70	Closed	SC
2	RHR Return to Suppression Pool (10-34A, B)		2	120	Closed	SC
2	RHR Drywell Spray (10-26A, B & 10-31A, B)		4	70	Closed	SC
2	RHR Suppression Chamber Spray (10-38A, B)		2	45	Closed	SC
3	Containment Air Compressor Suction (72-38A, B)		2	20	Open	GC
4	RHR Shutdown Cooling Supply (10-18, 10-17)	1	1	28	Closed	SC
	Standby Liquid Control Check Valves (11-16, 11-17)	1	1	NA	Closed	Process
*	Hydrogen Monitoring (109-75 A, 1-4; 109-75 B-D, 1-2) Sampling Valves - Inlet		10	NA	NA	NA
*	Hydrogen Monitoring (VG-24, 25, 33, 34)		4	NA	NA	NA

* These valves are remote manual sampling valves which do not receive an isolation signal. Only one valve in each line is required to be operable.

TABLE 4 7.2 NOTES

1. Isclation signals are as follows:

Grc p 1: The valves in Group 1 are closed upon any one of the following conditions:

1. Low-low reactor water level
2. High main steam line radiation
3. High main steam line flow
4. High main steam line tunnel temperature
5. Low main steam line pressure (run mode only)
6. Condenser low vacuum

Group 2: The valves in Group 2 are closed upon any one of the following conditions:

1. Low reactor water level
2. High drywell pressure

Group 3: The valves in Group 3 are closed upon any one of the following conditions:

1. Low reactor water level
2. High drywell pressure
3. High/low radiation - reactor building ventilation exhaust plenum or refueling floor

Group 4: The valves in Group 4 are closed upon any one of the following conditions:

1. Low reactor water level
2. High drywell pressure
3. High reactor pressure

Group 5: The valves in Group 5 are closed upon low reactor water level.

Group 6: The valves in Group 6 are closed upon any signal representing a steam line break in the HPCI system's or RCIC system's respective steam line. The signals indicating a steam line break for the respective steam line are as follows:

1. High steam line space temperature
2. High steam line flow
3. Low steam line pressure
4. High temperature in the main steam line tunnel (30 minute delay for the HPCI and the RCIC)

2. The closure time shall not be less than 3 seconds.

BASES:

3.7 STATION CONTAINMENT SYSTEMS

A. Primary Containment

The integrity of the primary containment and operation of the core standby cooling systems in combination limit the off-site doses to values less than to those suggested in 10 CFR 100 in the event of a break in the primary system piping. Thus, containment integrity is specified whenever the potential for violation of the primary reactor system integrity exists. Concern about such a violation exists whenever the reactor is critical, above atmospheric pressure and temperature above 212°F. An exception is made to this requirement during initial core loading and while a low power test program is being conducted and ready access to the reactor vessel is required. The reactor may be taken critical during the period; however, restrictive operating procedures will be in effect again to minimize the probability of an accident occurring. Procedures and the Rod Worth Minimizer would limit control worth to less than 1.30% delta k.

The pressure suppression pool water provides the heat sink for the reactor primary system energy release following postulated rupture of the system. The pressure suppression chamber water volume must absorb the associated decay and structural sensible heat released during primary system blowdown from 1000 psig.

Since all the gases in the drywell are purged into the pressure suppression chamber air space during a loss-of-coolant accident, the pressure resulting from isothermal compression plus the vapor pressure of the liquid must not exceed 62 psig, the allowable pressure suppression chamber pressure. The design volume of the suppression chamber (water and air) was obtained by considering that the total volume of reactor coolant to be condensed is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber (Reference Section 5.2 FSAR).

Using the minimum or maximum water volumes given in the specification, containment pressure during the design basis accident is approximately 44 psig, which is below the design of 56 psig.[3] The minimum volume of 68,000 ft^3 results in a submergency of approximately four feet. The majority of the Bodega tests[2] were run with a submerged length of four feet and with complete condensation. Thus, with respect to downcomer submergence, this specification is adequate.

The maximum temperature at the end of blowdown tested during the Humbolt Bay[1] and Bodega Bay tests was 170°F and this is conservatively taken to be the limit for complete condensation of the reactor coolant, although condensation would occur for temperature above 170°F.

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(1) Robbins, C. H., "Tests on a Full Scale 1/48 Segment of the Humbolt Bay Pressure Suppression Containment", GEAP-3596, November 17, 1960.

(2) Bodega Bay Preliminary Hazards Summary Report, Appendix 1, Docket 50-205, December 28, 1962.

(3) Code Allowable peak accident pressure is 62 psig.

BASES: 3.7 (Cont'd)

In conjunction with the Mark I Containment Long-Term Program, a plant unique analysis was performed (see Vermont Yankee letter, dated April 27, 1984, transmitting Teledyne Engineering Services Company Reports, TR-5319-1, Revision 2, dated November 30, 1983 and TR-5319-2, Revision 0) which demonstrated that all stresses in the suppression chamber structure, including shell, external supports, vent system, internal structures, and attached piping meet the structural acceptance criteria of NUREG-0661. The maintenance of a drywell-suppression chamber differential pressure of 1.7 psid and a suppression chamber water level corresponding to a downcomer submergence range of 4.29 to 4.54 ft. will assure the integrity of the suppression chamber when subjected to post-LOCA suppression pool hydrodynamic forces.

Using a 50°F rise (Section 5.2.4 FSAR) in the suppression chamber water temperature and a minimum water volume of 68,000 ft^3, the 170°F temperature which is used for complete condensation would be approached only if the suppression pool temperature is 120°F prior to the DBA-LOCA. Maintaining a pool temperature of 100°F will assure that the 170°F limit is not approached.

Experimental data indicate that excessive steam condensing loads can be avoided if the peak temperature of the suppression pool is maintained below 160°F during any period of relief valve operation with sonic conditions at the discharge exit. Specifications have been placed on the envelope of reactor operating conditions so that the reactor can be depressurized in a timely manner to avoid the regime of potentially high suppression chamber loadings.

In addition to the limits on temperature of the suppression chamber pool water, operating procedures define the action to be taken in the event a relief valve inadvertently opens or sticks open. This action would include: (1) use of all available means to close the valve, (2) initiate suppression pool water cooling heat exchangers, (3) initiate reactor shutdown, and (4) if other relief valves are used to depressurize the reactor, their discharge shall be separated from that of the stuck-open relief valve to assure mixing and uniformity of energy insertion to the pool.

Double isolation valves are provided on lines which penetrate the primary containment and open to the free space of the containment. Closure of one of the valves in each line would be sufficient to maintain the integrity of the pressure suppression system. Automatic initiation is required to minimize the potential leakage paths from the containment in the event of a loss-of-coolant accident. Details of the isolation valves are discussed in Section 5.2 of the FSAR.

The purpose of the vacuum relief valves is to equalize the pressure between the drywell and suppression chamber and suppression chamber and reactor building so that the structural integrity of the containment is maintained.

Technical Specification 3.7.A.9.c is based on the assumption that the operability testing of the pressure suppression chamber-reactor building vacuum breaker, when required, will normally be performed during the same four hour testing interval as the pressure suppression chamber-drywell vacuum breakers in order to minimize operation with <1.7 psi, differential pressure.

BASES: 3.7 (Cont'd)

The vacuum relief system from the pressure suppression chamber to Reactor Building consists of two 100% vacuum relief breakers (2 parallel sets of 2 valves in series). Operation of either system will maintain the pressure differential less than 2 psig; the external design pressure is 2 psig. With one vacuum breaker out of service there is no immediate threat to accident mitigation or primary containment and, therefore, reactor operation can be continued for 7 days while repairs are being made.

The capacity of the ten (10) drywell vacuum relief valves is sized to limit the pressure differential between the suppression chamber and drywell during post-accident drywell cooling operations to the design limit of 2 psig. They are sized on the basis of the Bodega Bay pressure suppression tests. The ASME Boiler and Pressure Vessel Code, Section III, Subsection B, for this vessel allows eight (8) operable valves, therefore, with two (2) valves secured, containment integrity is not impaired.

Each drywell-suppression chamber vacuum breaker is fitted with a redundant pair of limit switches to provide fail-safe signals to panel mounted indicators in the Reactor Building and alarms in the Control Room when the disks are open more than 0.050" at all points along the seal surface of the disk. These switches are capable of transmitting the disk closed to open signal with 0.01" movement of the switch plunger. Continued reactor operation with failed components is justified because of the redundance of components and circuits and, most importantly, the accessibility of the valve lever arm and position reference external to the valve. The fail safe feature of the alarm circuits assures operator attention if a line fault occurs.

The requirement to inert the containment is based on the recommendation of the Advisory Committee on Reactor Safeguards. This recommendation, in turn, is based on the assumption that several percent of the zirconium in the core will undergo a reaction with steam during the loss-of-coolant accident. This reaction would release sufficient hydrogen to result in a flammable concentration in the primary containment building. The oxygen concentration is therefore kept below 4% to minimize the possibility of hydrogen combustion.

General Electric has estimated that less than 0.1% of the zirconium would react with steam following a loss-of-coolant due to operation of emergency core cooling equipment. This quantity of zirconium would not liberate enough hydrogen to form a combustible mixture.

C. Standby Gas Treatment System and Secondary Containment System

The secondary containment is designed to minimize any ground level release of radioactive materials which might result from a serious accident. The Reactor Building provides secondary containment during reactor operation, when the drywell is sealed and in service; the Reactor Building provides primary containment when the reactor is shutdown and the drywell is open, as during refueling. Because the secondary containment is an integral part of the complete containment system, secondary containment is required at all times that primary containment is required except, however, for initial fuel loading and low power physics testing.

BASES: 3.7 (Cont'd)

The standby gas treatment system is designed to filter and exhaust the Reactor Building atmosphere to the stack during secondary containment isolation conditions, with a minimum release of radioactive materials from the Reactor Building to the environs. To insure that the standby gas treatment system will be effective in removing radioactive contaminates from the Reactor Building air, the system is tested periodically to meet the intent of ANSI N510-1975. Both standby gas treatment fans are designed to automatically start upon containment isolation and to maintain the Reactor Building pressure to approximately a negative 0.15 inch water gauge pressure; all leakage should be in-leakage. Should the fan fail to start, the redundant alternate fan and filter system is designed to start automatically. Each of the two fans has 100% capacity. This substantiates the availability of the operable circuit and results in no added risk; thus, reactor operation or refueling operation can continue. If neither circuit is operable, the plant is brought to a condition where the system is not required.

D. Primary Containment Isolation Valves

Double isolation valves are provided on lines that penetrate the primary containment and communicate directly with the reactor vessel and on lines that penetrate the primary containment and communicate with the primary containment free space. Closure of one of the valves in each line would be sufficient to maintain the integrity of the pressure suppression system. Automatic initiation is required to minimize the potential leakage paths from the containment in the event of a loss-of-coolant accident.

4.7 STATION CONTAINMENT SYSTEMS

A. Primary Containment System

The water in the suppression chamber is used only for cooling in the event of an accident, i.e., it is not used for normal operation; therefore, a weekly check of the temperature and volume is adequate to assure that adequate heat removal capability is present.

The interiors of the drywell and suppression chamber are painted to prevent rusting. The inspection of the paint during each major refueling outage, approximately once per year, assures the paint is intact. Experience with this type of paint at fossil fueled generating stations indicates that the inspection interval is adequate.

Because of the large volume and thermal capacity of the suppression pool, the volume and temperature normally changes very slowly and monitoring these parameters daily is sufficient to establish any temperature trends. By requiring the suppression pool temperature to be continually monitored and frequently logged during periods of significant heat addition, the temperature trends will be closely followed so that appropriate action can be taken. The requirement for an external visual examination following any event where potentially high loadings could occur provides assurance that no significant damage was encountered. Particular attention should be focused on structural discontinuities in the vicinity of the relief valve discharge since these are expected to be the points of highest stress. Visual inspection of the suppression chamber including water line regions each refueling outage is adequate to detect any changes in the suppression chamber structures.

BASES: 4.7 (Cont'd)

The primary containment preoperational test pressures are based upon the calculated primary containment pressure response in the event of a loss-of-coolant accident. The peak drywell pressure would be about 44 psig which would rapidly reduce to 27 psig within 10 seconds following the pipe break. Following the pipe break, the suppression chamber pressure rises to 27 psig within 10 seconds, equalizes with drywell pressure and therefore rapidly decays with the drywell pressure decay.[1]

The design pressure of the drywell and absorption chamber is 56 psig.[2] The design leak rate is 0.5%/day at a pressure of 62 psig. As pointed out above, the pressure response of the drywell and suppression chamber following an accident would be the same after about 10 seconds. Based on the primary containment pressure response and the fact that the drywell and suppression chamber function as a unit, the primary containment will be tested as a unit rather than the individual components separately.

The design basis loss-of-coolant accident was evaluated at the primary containment maximum allowable accident leak rate of 1.5%/day at 44 psig. The analysis showed that with this leak rate and a standby gas treatment system filter efficiency of 90% for halogens, 95% for particulates, and assuming the fission product release fractions stated in TID-14844, the maximum total whole body passing cloud dose is about 1.65 rem and the maximum total thyroid dose is about 280 rem at the site boundary over an exposure duration of two hours. The resultant dose that would occur for the duration of the accident at the low population distance of 5 miles is lower than those stated due to the variability of meteorological conditions that would be expected to occur over a 30-day period. Thus, these doses are the maximum that would be expected in the unlikely event of a design basis loss-of-coolant accident. These doses are also based on the assumption of no holdup in the secondary containment, resulting in a direct release of fission products from the primary containment through the filters and stack to the environs. Therefore, the specified primary containment leak rate and filter efficiency are conservative and provide margin between expected off-site doses and 10 CFR 100 guidelines. An additional factor of two for conservatism is added to the above doses by limiting the test leak rate (L a) to a value of 0.80%/day.

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(1) Section 5.2 of the FSAR.

(2) 62 psig is the maximum allowable peak accident pressure for this design (56 psig) pressure.

BASES: 4.7 (Cont'd)

The maximum allowable test leak rate at the peak accident pressure of 44 psig (La) is 0.80 weight % per day. The maximum allowable test leak rate at the retest pressure of 24 psig (Lt) has been conservatively determined to be 0.59 weight percent per day. This value will be verified to be conservative by actual primary containment leak rate measurements at both 44 psig and 24 psig upon completion of the containment structure.

To allow a margin for possible leakage deterioration between test intervals, the maximum allowable operational leak rate (Ltm), which will be met to remain on the normal test schedule, is 0.75 Lt.

As most leakage and deterioration of integrity is expected to occur through penetrations, especially those with resilient seals, a periodic leak rate test program of such penetration is conducted at the peak accident pressure of 44 psig to insure not only that the leakage remains acceptably low but also that the sealing materials can withstand the accident pressure.

The leak rate testing program is based on AEC guidelines for development of leak rate testing and surveillance schedules for reactor containment vessels.

Surveillance of the suppression Chamber-Reactor Building vacuum breakers consists of operability checks and leakage tests (conducted as part of the containment leak-tightness tests). These vacuum breakers are normally in the closed position and open only during tests or an accident condition. Operability testing is performed in conjunction with Specification 4.6.E. Inspections and calibrations are performed during the refueling outages; this frequency being based on equipment quality, experience, and engineering judgment.

The ten (10) drywell-suppression vacuum relief valves are designed to open to the full open position (the position that curtain area is equivalent to valve bore) with a force equivalent to a 0.5 psi differential acting on the suppression chamber face of the valve disk. This opening specification assures that the design limit of 2.0 psid between the drywell and external environment is not exceeded. Once each refueling outage each valve is tested to assure that it will open fully in response to a force less than that specified. Also it is inspected to assure that it closes freely and operates properly.

The containment design has been examined to establish the allowable bypass area between the drywell and suppression chamber as 0.12 ft^2. This is equivalent to one vacuum breaker open by three-eighths of an inch (3/8") as measured at all points around the circumference of the disk or three-fourths of an inch (3/4") as measured at the bottom of the disk when the top of the disk is on the seat. Since these valves open in a manner that is purely neither mode, a conservative allowance of one-half inch (1/2") has been selected as the maximum permissible valve opening. Assuming that permissible valve opening could be evenly divided among all ten vacuum breakers at once, valve open position assumed to indication for an individual valve must be activated less than fifty-thousandths of an inch (0.050") at all points along the seal surface of the disk. Valve closure within this limit may be determined by light indication from two independent position detection and indication systems. Either system provides a control room alarm for a nonseated valve.

BASES: 4.7 (Cont'd)

At the end of each refueling cycle, a leak rate test shall be performed to verify that significant leakage flow paths do not exist between the drywell and suppression chamber. The drywell pressure will be increased by at least 1 psi with respect to the suppression chamber pressure and held constant. The 2 psig set point will not be exceeded. The subsequent suppression chamber pressure transient (if any) will be monitored with a sensitive pressure guage. If the drywell pressure cannot be increased by 1 psi over the suppression chamber pressure it would be because a significant leakage path exists; in this event the leakage source will be identified and eliminated before power operation is resumed. If the drywell pressure can be increased by 1 psi over the suppression chamber the rate of change of the suppression chamber pressure must not exceed a rate equivalent to the rate of leakage from the drywell through a 1-inch orifice. In the event the rate of change exceeds this value then the source of leakage will be identified and eliminated before power operation is resumed.

The drywell-suppression chamber vacuum breakers are exercised in accordance with Specification 4.6.E and immediately following termination of discharge of steam into the suppression chamber. This monitoring of valve operability is intended to assure that valve operability and position indication system performance does not degrade between refueling inspections. When a vacuum breaker valve is exercised through an opening-closing cycle, the position indicating lights are designed to function as follows:

Full Closed (Closed to $\leq$0.050" open)	2 White - On
Open (>0.050" open to full open)	2 White - Off

During each refueling outage, two drywell-suppression chamber vacuum breakers will be inspected to assure sealing surfaces and components have not deteriorated. Since valve internals are designed for a 40-year lifetime, an inspection program which cycles through all valves in one-eighth of the design lifetime is extremely conservative.

Experience has shown that a weekly measurement of the oxygen concentration in the primary containment assures adequate surveillance of the primary containment atmosphere.

B. and C. Standby Gas Treatment System and Secondary Containment System

Initiating reactor building isolation and operation of the standby gas treatment system to maintain at least a 0.15 inch of water vacuum within the secondary containment provides an adequate test of the operation of the reactor building isolation valves, leakage tightness of the reactor building, and performance of the standby gas treatment system. Functionally testing of initiating sensors and associated trip channels demonstrates the capability for automatic actuation. Periodic testing gives sufficient confidence of reactor building integrity and standby gas treatment system performance capability.

BASES: 4.7 (Cont'd)

The test frequencies are adequate to detect equipment deterioration prior to significant defects, but the tests are not frequent enough to load the filters, thus reducing their reserve capacity too quickly. That the testing frequency is adequate to detect deterioration was demonstrated by the tests which showed no loss of filter efficiency after 2 years of operation in the rugged shipboard environment on the NS Savannah (ORNL 3726). Pressure drop tests across filter sections are performed to detect gross plugging of the filter media. Considering the relatively short time that the fans may be run for test purposes, plugging is unlikely, and the test interval is reasonable. Such heater tests will be conducted once during each operating cycle. Considering the simplicity of the heating circuit, the test frequency is sufficient. Air distribution tests will be conducted once during each operating cycle.

The in-place testing of charcoal filters is performed using a halogenated hydrocarbon, which is injected into the system upstream of the charcoal filters. Measurements of the challenge gas concentration upstream and downstream of the charcoal filters is made. The ratio of the inlet and outlet concentrations gives an overall indication of the leak tightness of the system. Although this is basically a leak test, since the filters have charcoal of known efficiency and holding capacity for elemental iodine and/or methyl iodine, the test also gives an indication of the relative efficiency of the installed system.

High-efficiency particulate air filters are installed before and after the charcoal filter to minimize potential release of particulates to the environment and to prevent clogging of the iodine filters. An efficiency of 99% is adequate to retain particulates that may be released to the Reactor Building following an accident. This will be demonstrated by testing with DOP as testing medium.

The efficiencies of the particulate and charcoal filters are sufficient to prevent exceeding 10CFR100 limits for the accidents analyzed. The analysis of post-accident hydrogen purge assumed a charcoal filter efficiency of 95%. Hence requiring in-place test efficiencies of 99% for these filters provides adequate margin. The laboratory methyl iodide removal test is performed at 95% relative humidity to assure adequate margin over the design relative humidity of 70%.

The test interval for filter efficiency was selected to minimize plugging of the filters. In addition, testing for methyl iodide removal efficiency will be demonstrated. This will be done either by removal of a charcoal sample cartridge which contains charcoal equivalent to the bed thickness or removing one adsorber tray from the system and using the charcoal therein, after mixing, to obtain at least two samples equivalent to the bed thickness. Any HEPA filters found defective should be replaced with filters qualified according to Regulatory Position C.3.d of Regulatory Guide 1.52. If laboratory test results are unacceptable, all charcoal adsorbent in the system should be replaced with charcoal adsorbent qualified according to Regulatory Guide 1.52.

BASES: 4.7 (Cont'd)

D. Primary Containment Isolation Valves

Those large pipes comprising a portion of the reactor coolant system whose failure could result in uncovering the reactor core are supplied with automatic isolation valves (except those lines needed for emergency core cooling system operation or containment cooling). The closure times specified herein and per Specification 4.6.E are adequate to prevent loss of more cooling from the circumferential rupture of any of these lines outside the containment than from a steam line rupture. Therefore, the isolation valve closure times are sufficient to prevent uncovering the core.

Purge and vent valve testing performed by Allis-Chalmers has demonstrated that all butterfly purge and vent valves installed at Vermont Yankee can close from full open conditions at design basis containment pressure. However, as an additional conservative measure, limit stops have been added to valves 16-19-7/7A, limiting the opening of these valves to 50° open while operating, as requested by NRC in their letter of May 22, 1984. (NVY 84-108)

In order to assure that the doses that may result from a steam line break do not exceed the 10CFR100 guidelines, it is necessary that no fuel rod perforation resulting from the accident occur prior to closure of the main steam line isolation valves. Analyses indicate the fuel rod cladding perforations would be avoided for the main steam valve closure times, including instrument delay, as long as 10.5 seconds. The test closure time limit of five seconds for these main steam isolation valves provides sufficient margin to assure that cladding perforations are avoided and 10CFR100 limits are not exceeded. Redundant valves in each line ensure that isolation will be effected applying the single failure criteria.

The main steam line isolation valves are functionally tested on a more frequent interval to establish a high degree of reliability.

The containment is penetrated by a large number of small diameter instrument lines. The flow check valves in these lines are tested for operability in accordance with Specification 4.6.E.

3.8 LIMITING CONDITIONS FOR OPERATION	4.8 SURVEILLANCE REQUIREMENTS

3.8 RADIOACTIVE EFFLUENTS

Applicability:

Applies to the release of all radioactive effluents from the plant.

Objective:

To assure that radioactive effluents are kept "as low as is reasonably achievable" in accordance with 10CFR50, Appendix I and, in any event, are within the limits specified in 10CFR20.

Specification:

A. Liquid Effluents: Concentration

1. The concentration of radioactive material in liquid effluents released from the site shall be limited to the concentrations specified in 10CFR Part 20, Appendix B, Table II, Column 2 for radionuclides other than noble gases and 2×10^{-4} uCi/ml total activity concentration for all dissolved or entrained noble gases.

2. With the concentration of radioactive material in liquid effluents released from the site exceeding the limits of Specification 3.8.A.1, immediately take action to decrease the release rate of radioactive materials and/or increase the dilution flow rate to restore the concentration to within the above limits.

4.8 RADIOACTIVE EFFLUENTS

Applicability:

Applies to the required surveillance of all radioactive effluents released from the plant.

Objective:

To ascertain that all radioactive effluents released from the plant are kept "as low as is reasonably achievable" in accordance with 10CFR50, Appendix I and, in any event, are within the limits specified in 10CFR20.

Specification

A. Liquid Effluents: Concentration

1. Radioactive material in liquid waste shall be sampled and analyzed in accordance with requirements of Table 4.8.1. The results of the analyses shall be used in accordance with the methods in the ODCM to assure that the concentrations at the point of release are limited to the values in Specification 3.8.A.1.

3.8 LIMITING CONDITIONS FOR OPERATION

B. Liquid Effluents: Dose

1. The dose or dose commitment to a member of the public from radioactive materials in liquid effluents released from the site shall be limited to the following:

   a. During any calendar quarter:

      less than or equal to 1.5 mrem to the total body, and

      less than or equal to 5 mrem to any organ, and

   b. During any calendar year:

      less than or equal to 3 mrem to the total body, and

      less than or equal to 10 mrem to any organ.

C. Liquid Radwaste Treatment

1. The liquid radwaste treatment system shall be used in its designed modes of operation to reduce the radioactive materials in the liquid waste prior to its discharge when the estimated doses due to the liquid effluent from the site, when averaged with all other liquid release over the last month, would exceed 0.06 mrem to the total body, or 0.2 mrem to any organ.

4.8 SURVEILLANCE REQUIREMENTS

B. Liquid Effluents: Dose

1. Cumulative dose contributions shall be determined in accordance with the methods in the ODCM at least once per month if releases during the period have occurred.

C. Liquid Radwaste Treatment

1. See Specification 4.8.B.1.

3.8 LIMITING CONDITIONS FOR OPERATION

D. Liquid Holdup Tanks

1. The quantity of radioactive material contained in any outside tank* shall be limited to less than or equal to 10 curies, excluding tritium and dissolved or entrained noble gases.

2. With the quantity of radioactive material in any outside tank* exceeding the limit of Specification 3.8.D.1, immediately take action to suspend all additions of radioactive material to the tank. Within 48 hours, reduce the tank contents to within the limit.

E. Gaseous Effluents: Dose Rate

1. The dose rate due to radioactive materials released in gaseous effluents from the site to areas at and beyond the site boundary shall be limited to the following:

   a. For noble gases; less than or equal to 500 mrem/yr to the total body and less than or equal to 3,000 mrem/yr to the skin, and

4.8 SURVEILLANCE REQUIREMENTS

D. Liquid Holdup Tanks

1. The quantity of radioactive material contained in each of the liquid holdup tanks* shall be determined to be within the limits of Specification 3.8.D.1 by analyzing a representative sample of the tank's contents within one week following the addition of radioactive materials to the tank. One sample may cover multiple additions.

E. Gaseous Effluents: Dose Rate

1. The dose rate due to noble gases in gaseous effluents shall be determined to be within the limits of Specification 3.8.E.1 in accordance with the methods in the ODCM.

2. The dose rate due to Iodine-131, Iodine-133, tritium and radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents shall be determined to be within the limits of Specification 3.8.E.1 in

*NOTE: Tanks included in this specification are only those outdoor tanks that are not surrounded by liners, dikes, or walls capable of holding the tank's contents, or that do not have tank overflows and surrounding area drains connected to the liquid radwaste treatment system.

3.8 LIMITING CONDITIONS FOR OPERATION

b. For Iodine-131, Iodine-133, tritium and radionuclides in particulate form with half-lives greater than 8 days; less than or equal to 1,500 mrem/yr to any organ.

2. With the dose rate(s) exceeding the above limits, immediately take action to decrease the release rate to within the limits of Specification 3.8.E.1.

F. <u>Gaseous Effluents: Dose from Noble Gases</u>

1. The air dose due to noble gases released in gaseous effluents from the site to areas at and beyond the site boundary shall be limited to the following:

a. During any calendar quarter:

less than or equal to 5 mrad for gamma radiation, and

less than or equal to 10 mrad for beta radiation, and

b. During any calendar year:

less than or equal to 10 mrad for gamma radiation, and

less than or equal to 20 mrad for beta radiation.

4.8 SURVEILLANCE REQUIREMENTS

accordance with the methods in the ODCM by obtaining representative samples and performing analyses in accordance with the sampling and analysis program specified in Table 4.8.2.

F. <u>Gaseous Effluents: Dose from Noble Gases</u>

1. Cumulative dose contributions for the total time period shall be determined in accordance with the methods in the ODCM at least once every month.

3.8 LIMITING CONDITIONS FOR OPERATION

G. Gaseous Effluents: Dose from Iodine-131, Iodine-133, Tritium, and Radionuclides in Particulate Form

1. The dose to a member of the public from Iodine-131, Iodine-133, tritium, and radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released from the site to areas at and beyond the site boundary shall be limited to the following:

   a. During any calendar quarter:

      less than or equal to 7.5 mrem to any organ, and

   b. During any calendar year:

      less than or equal to 15 mrem to any organ.

H. Gaseous Radwaste Treatment

1. The Augmented Off-Gas System (AOG) shall be used in its designed mode of operation to reduce noble gases in gaseous waste prior to their discharge whenever the main condenser steam jet air ejector (SJAE) is in operation.

I. Ventilation Exhaust Treatment

1. The AOG and Radwaste Building Ventilation Filter (HEPA) Systems shall be used to reduce particulate materials in gaseous waste prior to their discharge from those buildings when the estimated doses due to gaseous effluent releases from the site

4.8 SURVEILLANCE REQUIREMENTS

G. Gaseous Effluents: Dose from Iodine-131, Iodine-133, Tritium, and Radionuclides in Particulate Form

1. Cumulative dose contributions for the total time period shall be determined in accordance with the methods in the ODCM at least once every month.

H. Gaseous Radwaste Treatment

1. The readings of the relevant instrument shall be checked every 12 hours when the main condenser SJAE is in use to ensure that the AOG is functioning.

I. Ventilation Exhaust Treatment

1. See Specification 4.8 F.1 for surveillance related to AOG and Radwaste Building ventilation filter system operation.

3.8 LIMITING CONDITIONS FOR OPERATION

to areas at and beyond the site boundary would exceed 0.3 mrem to any organ over one month.

J. Explosive Gas Mixture

1. If the hydrogen concentration in the off-gas downstream of the operating recombiner reaches four percent, take appropriate action that will restore the concentration to within the limit within 48 hours.

K. Steam Jet Air Ejector (SJAE)

1. Gross radioactivity release rate from the SJAE shall be limited to less than or equal to 0.16 Ci/sec (after 30 minutes decay).

2. With the gross radioactivity release rate at the SJAE exceeding the above limit, restore the gross radioactivity release rate to within its limit within 72 hours or be in at least Hot Standby within the subsequent 12 hours.

3. With the gross radioactivity release rate at the SJAE greater than or equal to 1.5 Ci/sec (after 30-minute decay), restore the gross radioactivity release rate to less than 1.5 Ci/sec (after 30-minute decay), or be in Hot Standby within 12 hours.

4.8 SURVEILLANCE REQUIREMENTS

J. Explosive Gas Mixture

1. The concentration of hydrogen in the off-gas system downstream of the recombiners shall be continuously monitored by the hydrogen monitor required operable by Table 3.9.2.

K. Steam Jet Air Ejector (SJAE)

1. The gross radioactivity release rate shall be continuously monitored in accordance with Specification 3.9.B.

2. The gross radioactivity release rate of noble gases from the SJAE shall be determined to be within the limit of Specification 3.8.K.1 at the following frequencies by performing an isotopic analysis (for Xe-138, Xe-135, Xe-133, Kr-88, Kr-85m, Kr-87) on a representative sample of gases taken at the discharge.

   a. Once per week.
   b. Within 4 hours following an increase of 25% or 5000 microcuries/sec, whichever is greater, in steady-state activity levels during steady-state reactor operation, as indicated by the SJAE monitor.

3.8 LIMITING CONDITIONS FOR OPERATION

L. Primary Containment

1. If the primary containment is to be Vented/Purged, it shall be Vented/Purged through the Standby Gas Treatment System whenever the airborne radioactivity levels in containment exceed the levels specified in 10CFR20, Appendix B, Table I, Column 1 and notes 1-5 thereto.

2. With the requirements of Specification 3.8.L.1 not satisfied, immediately suspend all Venting/Purging of the containment.

3. During normal refueling and maintenance outages when primary containment is no longer required, then Specification 3.8.G shall supersede Specifications 3.8.L.1 and 2.

M. Total Dose

1. The dose or dose commitment to a member of the public* from all station sources is limited to less than or equal to 25 mrem to the total body or any organ (except the thyroid, which is limited to less than or equal to 75 mrem) over a calendar year.

4.8 SURVEILLANCE REQUIREMENTS

L. Primary Containment

1. The primary containment shall be sampled prior to venting/purging per Table 4.8.2, and if the results indicate radioactivity levels in excess of the limits of Specification 3.8.L.1, the containment shall be aligned for venting/purging through the Standby Gas Treatment System. No sampling shall be required if the venting/purging is through the Standby Gas Treatment (SBGT) System.

M. Total Dose

1. Cumulative dose contributions from liquid and gaseous effluents shall be determined in accordance with Specifications 4.8.B.1, 4.8.F.1, and 4.8.G.1.

*NOTE: For this Specification a member of the public may be taken as a real individual accounting for his actual activities.

3.8 LIMITING CONDITIONS FOR OPERATION

2. With the calculated dose from the release of radioactive materials in liquid or gaseous effluents exceeding twice the limits of Specifications 3.8.B.1.a, 3.8.B.1.b, 3.8.F.1.a, 3.8.F.1.b, 3.8.G.1.a, or 3.8.G.1.b, calculations should be made, including direct radiation contributions from the station to determine whether the above limits of Specification 3.8.M.1 have been exceeded.

N. Solid Radioactive Waste

1. The solid radwaste system shall be used in accordance with a Process Control Program as described in Section 6.12 to process wet radioactive waste (spent resins/filter sludges) to meet shipping and burial ground requirements.

2. With the provisions of Specification 3.8.N.1 not satisfied, suspend shipments of defectively processed or defectively packaged solidified wet radioactive wastes from the site.

4.8 SURVEILLANCE REQUIREMENTS

2. Cumulative dose contributions from direct radiation from plant sources shall be determined in accordance with the methods in the ODCM. This requirement is applicable only under conditions set forth in Specification 3.8.M.2.

N. Solid Radioactive Waste

1. Verification of solidification of wet waste shall be performed as required and in accordance with the Process Control Program.

TABLE 4.8.1

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

Liquid Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) (uCi/ml)a
Batch Waste Release Tanks[b]	Prior to each release Each Batch	Prior to each release Each Batch	Principal Gamma Emitters[d]	5×10^{-7}
			I-131	1×10^{-6}
	One Batch per month sampled prior to a release	Once per month	Dissolved and Entrained Gases (Gamma Emitters)	1×10^{-5}
	Prior to each release Each Batch	Once per month Composite[c]	H-3	1×10^{-5}
			Gross Alpha	1×10^{-7}
	Prior to each release Each Batch	Once per quarter Composite[c]	Sr-89, Sr-90	5×10^{-8}
			Fe-55	1×10^{-6}

TABLE 4.8.1 NOTES:

a. The LLD is 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):

$$LLD = \frac{4.66 * S_b}{E * V * K * Y * e^{-\lambda * \Delta t}}$$

where:

LLD = the lower limit of detection as defined above (microcuries or picocuries/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/minute)

E = the counting efficiency (counts/disintegration)

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

K = 2.22×10^6 disintegrations/minute/microcurie or 2.22 disintegration/minute/picocurie as applicable

Y = the fractional radiochemical yield (when applicable)

λ = the radioactive decay constant for the particular radionuclide (/minute)

Δt = the elapsed time between sample collection and analysis (minutes)

Typical values of E, V, Y and Δt can be used in the calculation. In calculating the LLD for a radionuclide determined by gamma-ray spectrometry, the background shall include the typical contributions of other radionuclides normally present in the samples.

Analysis shall be performed in such a manner that the stated LLDs will be achieved under routine conditions. Occasionally, background fluctuations, unavoidably small sample sizes, the presence of interfering nuclides, or other uncontrollable circumstances may render these LLDs unavailable.

It should be recognized that the LLD is defined as a "before the fact" limit representing the capability of a measurement system and not as an "after the fact" limit for a particular measurement. This does not preclude the calculation of an "after the fact" LLD for a particular measurement based upon the actual parameters for the sample in question and appropriate decay correction parameters such as decay while sampling and during analysis.

b. A batch release is the discharge of liquid wastes of a discrete volume. Prior to sampling for analysis, each batch shall be isolated and then thoroughly mixed to assure representative sampling.

TABLE 4.8.1 NOTES: (Cont'd)

c. 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 which is representative of the liquids released. 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.

d. The principal gamma emitters for which the LLD specification will apply are exclusively 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 detected and reported. Other peaks which are measurable and identifiable, together with the above nuclides, shall also be identified and reported. Nuclides which are below the LLD for the analyses should not be reported as being present at the LLD level, but as "not detected". When unusual circumstances result in LLDs higher than required, the reasons shall be documented in the Semiannual Effluent Release Report.

TABLE 4.8.2

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

Gaseous Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) (uCi/ml)a
A. Steam Jet Air Ejector	Once per week Grab Sample	Once per week	Xe-138, Xe-135, Xe-133, Kr-88, Kr-87, Kr-85M	1×10^{-4}
B. Containment Purge	Prior to each release Each Purge Grab Sample	Prior to each release Each Purge	Principal Gamma Emitters[d]	1×10^{-4}
C. Main Plant Stack	Once per month[c] Grab Sample	Once per month[c]	Principal Gamma Emitters[d]	1×10^{-4}
			H-3	1×10^{-6}
	Continuous[e]	Once per week[b] Charcoal Sample	I-131[f]	1×10^{-12}
	Continuous[e]	Once per week[b] Particulate Sample	Principal Gamma Emitters[d] (I-131)	1×10^{-11}
	Continuous[e]	Once per month Composite Particulate Sample	Gross Alpha	1×10^{-11}
	Continuous[e]	Once per quarter Composite Particulate Sample	Sr-89, Sr-90	1×10^{-11}
	Continuous	Noble Gas Monitor	Noble Gases Gross Beta or Gamma	1×10^{-5}

TABLE 4.8.2 NOTES

a. See footnote a. of Table 4.8.1.

b. Samples shall be changed at least once per 7 days and analyses shall be completed within 48 hours after removal from samplers. 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 25% of rated thermal power in one hour, and analyses shall be completed within 48 hours of changing the samples. When samples collected for 24 hours are analyzed, the corresponding LLDs may be increased by a factor of 10. This requirement to sample at least once per 24 hours for 7 days applies only if: (1) analysis shows that the dose equivalent I-131 concentration in the primary coolant has increased more than a factor of 3 and the resultant concentration is at least 1×10^{-1} μCi/ml; and (2) the noble gas monitor shows that effluent activity has increased more than a factor of 3.

c. Sampling and analyses shall also be performed following shutdown, startup, or a thermal power change exceeding 25% of rated thermal power per hour unless: (a) analysis shows that the dose equivalent I-131 concentration in the primary coolant has not increased more than a factor of 3 and the resultant concentration is at least 1×10^{-1} μCi/ml; and (2) the noble gas monitor shows that effluent activity has not increased more than a factor of 3.

d. The principal gamma emitters for which the LLD specification will apply are exclusively the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135 and Xe-138 for gaseous emissions, and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141 and Ce-144 for particulate emissions. This list does not mean that only these nuclides are to be detected and reported. Other peaks which are measurable and identifiable, together with the above nuclides, shall also be identified and reported. Nuclides which are below LLD for the analyses should not be reported as being present at the LLD level for that nuclide, but as "not detected". When unusual circumstances result in LLDs higher than required, the reasons shall be documented in the Semiannual Effluent Release Report.

e. 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 3.8.E.1, 3.8.F.1 and 3.8.G.1.

f. The gaseous waste sampling and analysis program does not explicitly require sampling and analysis at a specified LLD to determine the I-133 release. Estimates of I-133 releases shall be determined by counting the weekly charcoal sample for I-133 (as well as I-131) and assume a constant release rate for the release period.

BASES:

3.8 RADIOACTIVE EFFLUENTS

A. Liquid Effluents: Concentration

This specification is provided to ensure that the concentration of radioactive materials released in liquid waste effluents from the site above background (at the point of discharge from the plant discharge into Connecticut River) will be less than the concentration levels specified in 10CFR Part 20 Appendix B, Table II, Column 2. This limitation provides additional assurance that the levels of radioactive materials in bodies of water outside the site will result in exposure within (1) the Section II.A design objectives of Appendix I, 10CFR Part 50, to a member of the public, and (2) the limits of 10CFR Part 20.106 (e) to the population.

The concentration limit for noble gases is based upon the assumption that Xe-135 is the controlling radionuclide and its MPC in air (submersion) was converted to an equivalent concentration in water using the International Commission on Radiological Protection (ICRP) Publication 2.

B. Liquid Effluents: Dose

This specification is provided to implement the requirements of Sections II.A, III.A, and IV.A of Appendix I, 10CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.A of Appendix I. The requirements provide 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 will be kept "as low as is reasonably achievable". The Surveillance Requirements implement the requirements in Section III.A of Appendix I, i.e., that conformance with the guides of Appendix I be shown by calculational procedures based on models and data such that the actual exposure of an individual through appropriate pathways is unlikely to be substantially underestimated. In addition, there is reasonable assurance that the operation of the facility will not result in radionuclide concentrations in potable drinking water that are in excess of the requirements of 40CFR 141. No drinking water supplies drawn from the Connecticut River below the plant have been identified. The appropriate dose equations for implementation through requirements of the Specification are described in the Vermont Yankee Off-Site Dose Calculation Manual. The equations specified in the ODCM for calculating the doses due to the actual release rates of radioactive materials in liquid effluents were developed from 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 10CFR Part 50, Appendix I", Revision 1, October 1977 and Regulatory Guide 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I", Revision 1, April 1977.

C. Liquid Radwaste Treatment

The requirement that the appropriate portions of this system as indicated in the ODCM be used, when specified, provides assurance that the releases of radioactive materials in liquid effluents will be kept "as low as is reasonably achievable". This specification implements the requirements of 10CFR Part 50.36a and the design objective given in Section II.D of Appendix I to 10CFR Part 50. The specified limits governing the use of appropriate portions of the

BASES: 3.8 (Cont'd)

liquid radwaste treatment system were specified as a suitable fraction of the dose design objectives set forth in Section II.A of Appendix I, 10CFR Part 50, for liquid effluents.

D. Liquid Holdup Tanks

The tanks listed in this Specification include all outdoor tanks that contain radioactivity that are not surrounded by liners, dikes, or walls capable of holding the tank contents, or that do not have tank overflows and surrounding area drains connected to the liquid radwaste treatment system.

Restricting the quantity of radioactive material contained in the specified tanks provides assurance that in the event of an uncontrolled release of the tanks' contents, the resulting concentrations would be less than the limits of 10CFR Part 20, Appendix B, Table II, Column 2, at the nearest portable water supply and the nearest surface water supply in an unrestricted area.

E. Gaseous Effluents: Dose Rate

This specification is provided to ensure that the dose at any time at and beyond the site boundary from gaseous effluents will be within the annual dose limits of 10CFR Part 20. The annual dose limits are the doses associated with the concentrations of 10CFR Part 20, Appendix B, Table II, Column 1. These limits provide reasonable assurance that radioactive material discharged in gaseous effluents will not result in the exposure of member(s) of the public either within or outside the site boundary, to annual average concentrations exceeding the limits specified in Appendix B, Table II of 10CFR Part 20 [10 CFR Part 20.106(b)]. For member(s) of the public who may at times be within the site boundary, the occupancy of the individual will be sufficiently low to compensate for any increase in the atmospheric diffusion factor above that for the site boundary. The specified limits as determined by the methodology in the ODCM, restrict, at all times, the corresponding gamma and beta dose rates above background to a member of the public at or beyond the site boundary to (500) mrem/year to the total body or to (3,000) mrem/year to the skin.

Specification 3.8.E.b also restricts, at all times, comparable with the length of the sampling periods of Table 4.8.2 the corresponding thyroid dose rate above background to an infant via the cow-milk-infant pathway to 1500 mrem/year for the nearest cow to the plant.

F. Gaseous Effluents: Dose from Noble Gases

This specification is provided to implement the requirements of Sections II.B, III.A, and IV.A of Appendix I, 10CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.B of Appendix I. The requirements provide 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 gaseous effluents will be kept "as low as is reasonably achievable". The Surveillance Requirements implement the requirements in Section III.A of Appendix I, i.e., that conformance with the guides of Appendix I be shown by calculational procedures based on models and data such that the actual exposure of any member of the public through appropriate pathways is unlikely to be substantially underestimated. The appropriate dose equations are

BASES: 3.8 (Cont'd)

specified in the ODCM for calculating the doses due to the actual releases of radioactive noble gases in gaseous effluents. The ODCM also provides for determining the air doses at the site boundary based upon the historical average atmospheric conditions.

The equations specified in the ODCM for calculating the doses due to the actual release rates of radioactive noble gases in gaseous effluents were developed from 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 10CFR 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.

G. Gaseous Effluents: Dose from Iodine-131, Iodine-133, Tritium, and Radionuclides in Particulate Form

This specification is provided to implement the requirements of Section II.C, III.A, and IV.A of Appendix I, 10CFR Part 50. The Limiting Condition for Operation are the guides set forth in Section II.C of Appendix I. The requirements provide 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 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 models and data such that the actual exposure of a member of the public through appropriate pathways is unlikely to be substantially underestimated. The equations specified in the ODCM for calculating the doses due to the actual release rates of the subject materials were also developed using 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 10CFR 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, Iodine-133, tritium, and radionuclides in particulate form with half-lives greater than 8 days are dependent on the existing radionuclide pathways to man, in areas at and beyond its site boundary. The pathways which were examined in the development of these specifications 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 man.

H. Gaseous Radwaste Treatment

The requirement that the appropriate portions of the Augmented Off-Gas (AOG) System be used whenever the SJAE is in operation provides reasonable assurance that the releases of radioactive materials in gaseous effluents will be kept "a low as is reasonably achievable". This specification implements the requirements of 10CFR Part 50.36a and the design objectives of Appendix I to 10CFR Part 50.

BASES: 3.8 (Cont'd)

I. Ventilation Exhaust Treatment

The requirement that the AOG Building and Radwaste Building HEPA filters be used when specified provides reasonable assurance that the release of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable". This specification implements the requirements of 10CFR Part 50.36a and the design objective of Appendix I to 10CFR Part 50. The requirements governing the use of the appropriate portions of the gaseous radwaste filter systems were specified by the NRC in NUREG-0473, Revision 2 (July 1979) as a suitable fraction of the guide set forth in Sections II.B and II.C of Appendix I, 10CFR Part 50, for gaseous effluents.

J. Explosive Gas Mixture

The hydrogen monitors are used to detect possible hydrogen buildups which could result in a possible hydrogen explosion. Automatic isolation of the off-gas flow would prevent the hydrogen explosion and possible damage to the augmented off-gas system. Maintaining the concentration of hydrogen below its flammability limit provides assurance that the releases of radioactive materials will be controlled.

K. Steam Jet Air Ejector (SJAE)

Restricting the gross radioactivity release rate of gases from the main condenser SJAE provides reasonable assurance that the total body exposure to an individual at the exclusion area boundary will not exceed a small fraction of the limits of 10CFR Part 100 in the event this effluent is inadvertently discharged directly to the environment without treatment. This specification implements the requirements of General Design Criteria 60 and 64 of Appendix A to 10CFR Part 50.

L. Primary Containment (MARK I)

This specification provides reasonable assurance that releases from containment purging/venting operations will be filtered through the Standby Gas Treatment System so that the annual dose limits of 10CFR Part 20 at the site boundary will not be exceeded. The dose objectives of Specification 3.8.G restrict purge/venting operations when the Standby Gas Treatment System is not in use and gives reasonable assurance that all releases from the plant will be kept "as low as is reasonably achievable".

M. Total Dose

This specification is provided to meet the dose limitations of 40CFR Part 190 that have been incorporated into 10CFR Part 20 by 46 FR 18525. The specification requires the preparation and submittal of a Specific Report whenever the calculated doses from plant radioactive effluents exceed twice the design objective doses of Appendix I. For sites containing up to 4 reactors, it is highly unlikely that the resultant dose to a member of the public will exceed the dose limits of 40CFR Part 190 if the individual reactors remain within the reporting requirement level. The Special Report will describe a course of action that should result in the limitation of the annual dose to a member of the public to within the 40CFR Part 190 limits. For the purposes of the Special Report, it may be assumed that the dose commitment to the member of the public is estimated to exceed the requirements of 40CFR Part 190, the Special Report with a request for a variance (provided the release conditions resulting in violation of 40CFR Part 190 have not already been

BASES: 3.8 (Cont'd)

corrected), in accordance with the provisions of 40CFR Part 190.11 and 10CFR Part 20.405c, is considered to be a timely request and fulfills the requirements of 40CFR Part 190 until NRC staff action is completed. The variance only relates to the limits of 40CFR Part 190, and does not apply in any way to the other requirements for dose limitation of 10CFR Part 20, as addressed in Specification 3.8.A and 3.8.E. An individual is not considered a member of the public during any period in which he/she is engaged in carrying out any operation that is part of the nuclear fuel cycle.

N. Solid Radioactive Waste

This specification implements the requirements of 10CFR Part 50.36a with respect to the handling of solid radioactive waste (spent resin and filter sludges only). The establishment and implementation of a Process Control Program (PCP), provides the operational guidelines by which proper dewatering of filter media and spent resins in preparation for off-site disposal is assured.

3.9 LIMITING CONDITIONS FOR OPERATION	4.9 SURVEILLANCE REQUIREMENTS

3.9 RADIOACTIVE EFFLUENT MONITORING SYSTEMS

Applicability:

Applies to the monitoring systems or programs which perform a surveillance, protective or controlling function on the release of radioactive effluents from the plant and their identification in the environment.

Objective:

To assure the operability of the radioactive effluent monitoring systems and environmental programs.

Specifications:

A. Liquid Effluent Instrumentation

1. During periods of release through the monitored pathway, the radioactive liquid effluent monitoring instrumentation channel shall be operable in accordance with Table 3.9.1 with their alarm setpoints set to ensure that the limits of Specification 3.8.A.1 are not exceeded.

B. Gaseous Effluent Instrumentation

1. The gaseous process and effluent monitoring instrumentation channels shall be operable in accordance with Table 3.9.2 with their alarm/trip setpoints set to ensure that the limits of Specifications 3.8.E.1.a, 3.8.J.1, and 3.8.K.1 are not exceeded.

4.9 RADIOACTIVE EFFLUENT MONITORING SYSTEMS

Applicability:

Applies to the required surveillance of the monitoring systems or programs which perform a surveillance, protective or controlling function on the release of radioactive effluents from the plant and their identification in the environment.

Objective:

To specify the type and frequency of surveillance to be applied to the radioactive effluent monitoring system and environmental programs.

Specifications:

A. Liquid Effluent Instrumentation

1. Each radioactive liquid effluent monitoring instrumentation channel shall be tested and calibrated as indicated in Table 4.9.1.

B. Gaseous Effluent Instrumentation

1. Each gaseous process or effluent monitoring instrumentation channel shall be tested and calibrated as indicated in Table 4.9.2.

3.9 LIMITING CONDITIONS FOR OPERATION

C. Radiological Environmental Monitoring Program

1. The radiological environmental monitoring program shall be conducted as specified in Table 3.9.3.

D. Land Use Census

1. A land use census shall be conducted to identify the location of the nearest milk animal and the nearest residence in each of the 16 meteorological sectors within a distance of five miles. The survey shall also identify the nearest milk animal (within 3 miles of the plant) to the point of predicted highest annual average D/Q value in each of the three major meteorological sectors due to elevated releases from the plant stack.

2. With a land use census identifying one or more locations which yield a calculated dose or dose commitment (via the same exposure pathway) at least 20 percent greater than at a location from which samples are currently being obtained in accordance with Specification 3.9.C.1, add the new location(s) to the radiological environmental monitoring program within 30 days if permission from the owner to collect samples can be obtained, and sufficient sample volume is available. The sampling location(s), excluding the control

4.9 SURVEILLANCE REQUIREMENTS

C. Radiological Environmental Monitoring Program

1. The radiological environmental monitoring samples shall be collected pursuant to Table 3.9.3 from the locations given in the ODCM and shall be analyzed pursuant to the requirements of Table 3.9.3 and the detection capabilities required by Table 4.9.3.

D. Land Use Census

1. The land use census shall be conducted at least once per year between the dates of June 1 and October 1 by either a door-to-door survey, aerial survey, or by consulting local agricultural authorities. The results of the land use census shall be included in the annual Radiological Environmental Surveillance Report pursuant to Specification 6.7.C.3.

3.9 LIMITING CONDITIONS FOR OPERATION

station location, having the lowest calculated dose or dose commitment (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.

E. Intercomparison Program

1. Analyses shall be performed on referenced radioactive materials supplied as part of an Intercomparison Program which has been approved by NRC.

4.9 SURVEILLANCE REQUIREMENTS

E. Intercomparison Program

1. A summary of the results of analyses performed as part of the above required Intercomparison Program shall be included in the Annual Radiological Environmental Surveillance Report. The identification of the NRC approved Intercomparison Program which is being participated in shall be stated in the ODCM.

TABLE 3.9.1

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

	Minimum Channels Operable	Notes
1. Gross Radioactivity Monitors not Providing Automatic Termination of Release		
a. Liquid Radwaste Discharge Monitor	1*	1,4,5
b. Service Water Discharge Monitor	1	2,4,5
2. Flow Rate Measurement Devices		
a. Liquid Radwaste Discharge Flow Rate Monitor	1*	3,4

* During releases via this pathway.

TABLE 3.9.1 NOTES

NOTE 1 - With the number of channels operable less than required by the minimum channels operable requirement, effluent releases may continue provided that prior to initiating a release:

a. At least two independent samples are analyzed in accordance with Specification 4.8.A.1, and

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

Otherwise, suspend release of radioactive effluents via this pathway.

NOTE 2 - With the number of channels operable less than required by the minimum channels operable requirement, effluent releases via this pathway may continue provided that, at least once per 24 hours, grab samples are collected and analyzed for gross radioactivity (beta or gamma) at a lower limit of detection of at least 10^{-7} microcurie/ml.

NOTE 3 - With the number of channels operable less than required by the minimum channels operable requirement, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 4 hours during actual releases. Pump performance curves may be used to estimate flow.

NOTE 4 - With the number of channels operable less than required by the minimum channels operable requirement, exert reasonable efforts to return the instrument(s) to operable status prior to the next release.

NOTE 5 - The alarm setpoints of these channels shall be determined and adjusted in accordance with the methodology and parameters in the Off-Site Dose Calculation Manual (ODCM). With a radioactive liquid effluent monitoring instrumentation channel alarm setpoint less conservative than a value which will ensure that the limits of e.8.A.1 are met during periods of release, immediately take action to suspend the release of radioactive liquid effluents monitored by the affected channel or declare the channel inoperable; or change the setpoint so it is acceptably conservative.

TABLE 3.9.2

GASEOUS EFFLUENT MONITORING INSTRUMENTATION

Instrument	Minimum Channels Operable	Notes
1. Steam Jet Air Ejector (SJAE)		
a. Noble Gas Activity Monitor	1	7, 8, 9
2. Augmented Off-Gas System		
a. Noble Gas Activity Monitor Between the Charcoal Bed System and the Plant Stack (Providing Alarm and Automatic Termination of Release)	1	2, 5, 6, 7
b. Flow Rate Monitor	1	1, 5, 6
c. Hydrogen Monitor	1	3, 5, 6
3. Plant Stack		
a. Noble Gas Activity Monitor	1	5, 7, 10
b. Iodine Sampler Cartridge	1	4, 5
c. Particulate Sampler Filter	1	4, 5
d. Sampler Flow Integrator	1	1, 5
e. Stack Flow Rate Monitor	1	1, 5

TABLE 3.9.2 NOTES

NOTE 1 - With the number of channels operable less than required by the minimum channels operable requirement, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 4 hours.

NOTE 2 - With the number of channels operable less than required by the minimum channels operable requirement, effluent releases via this pathway may continue for a period of up to 7 days provided that at least one of the stack monitoring systems is operable and off-gas system temperature and pressure are measured continuously.

NOTE 3 - With the number of channels operable less than required by the minimum channels operable requirement, operation of the AOG System may continue provided gas samples are collected at least once per 24 hours and analyzed within the following 4 hours, or an orderly transfer of the off-gas effluents from the operating recombiner to the standby recombiner shall be made.

NOTE 4 - With the number of channels operable less than required by the minimum channels operable requirement, effluent releases via the affected pathway may continue provided samples are continuously collected with auxiliary sampling equipment.

NOTE 5 - With the number of channels operable less than required by the minimum channels operable requirement, exert reasonable efforts to return the instrument(s) to operable status within 30 days.

NOTE 6 - During releases via this pathway.

NOTE 7 - The alarm/trip setpoints of these channels shall be determined and adjusted in accordance with the methodology and parameters in the Off-Site Dose Calculation Manual (ODCM). With a gaseous process or effluent monitoring instrumentation channel alarm/trip setpoint less conservative than a value which will ensure that the limits of 3.8.E.1.a and 3.8.K.1 are met, immediately take actions to suspend the release of radioactive gaseous effluents monitored by the affected channel, or declare the channel inoperable, or change the setpoint so it is acceptably conservative.

NOTE 8 - Minimum channels operable required only during operation of the Steam Jet Air Ejector.

NOTE 9 - With the number of channels operable less than required by the minimum channels operable requirement, gases from the SJAE may be released to the environment for up to 72 hours provided:

1. The AOG System is not bypassed; and

2. The AOG System noble gas activity monitor is operable.

NOTE 10 - With the number of channels operable less than required by the minimum channels operable requirement, effluent releases via this pathway may continue provided grab samples are taken at least once per 12 hours and these samples are analyzed for gross activity within 24 hours.

TABLE 3.9.3

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Sample Locations[a]	Sampling and Collection Frequency	Type and Frequency of Analysis
1. AIRBORNE			
a. Radioiodine and Particulates	Samples from 5 locations: 1 sample from up valley, within 4 miles of Site Boundary. (major wind direction) 1 sample from down valley, within 4 miles of Site Boundary. (major wind direction) 1 sample each from the vicinity of two nearby communities, within 10 miles of Site Boundary. 1 sample from a control location.	Continuous operation of sampler with sample collection semimonthly or more frequently as required by dust loading or plant effluent releases.[h]	Radioiodine canister: Analyze each sample for I-131. Particulate sampler: Gross beta radioactivity analysis on each sample following filter change.[c] Composite (by location) for gamma isotopic[d] at least once per quarter.

TABLE 3.9.3
(Cont'd)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Sample Locations[a]	Sampling and Collection Frequency	Type and Frequency of Analysis
2. DIRECT RADIATION[b]	40 routine monitoring stations as follows: 16 incident response stations (one in each meteorological sector) within a range of 0 to 4 km[8]; 16 incident response stations (one in each meteorological sector) within a range of 2 to 8 km[8]; the balance of the stations to be placed in special interest areas and control station areas.	Quarterly.	Gamma dose, at least once per quarter. Incident response TLDs in the outer monitoring locations, de-dose only quarterly unless gaseous release LCO was exceeded in period.

TABLE 3.9.3
(Cont'd)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Sample Locations[a]	Sampling and Collection Frequency	Type and Frequency of Analysis
3. WATERBORNE			
a. Surface[e]	1 sample upstream.	Monthly grab sample.	Gamma isotopic analysis[d] of each sample. Tritium analysis of composite sample at least once per quarter.
	1 sample downstream.	Composite sample collected over a period of one month[f].	
b. Ground	1 sample from within 8 km distance.	Quarterly.	Gamma isotopic[d] and tritium analyses of each sample.
	1 sample from a control location.	Quarterly.	
c. Sediment from Shoreline	1 sample from downstream area with existing or potential recreational value.	Semiannually.	Gamma isotopic analysis[d] of each sample.
	1 sample from north storm drain outfall.	As specified in the ODCM.	

TABLE 3.9.3
(Cont'd)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Sample Locations[a]	Sampling and Collection Frequency	Type and Frequency of Analysis
4. INGESTION			
a. Milk	Samples from milking animals in 3 locations within 5 km distance having the highest dose potential. If there are less than 3 primary locations available then 1 or more secondary sample from milking animals in each of 3 areas between 5 and 8 km distance where doses are calculated to be greater than 1 mrem per year.	Semimonthly if milking animals are identified on pasture; at least once per month at other times.	Gamma isotopic[d] and I-131 analysis of each sample.
	1 sample from milking animals in a control location.		
b. Fish	1 sample of two recreationally important species in vicinity of plant discharge area.	Semiannually.	Gamma isotopic analysis[d] on edible portions.
	1 sample (preferably of same species) in areas not influenced by plant discharge.		
c. Vegetation	1 grass sample at each air sampling station.	Quarterly when available.	Gamma isotopic analysis[d] of each sample.
	1 silage sample at each milk sampling station (as available).	At time of harvest.	Gamma isotopic analysis[d] of each sample.

TABLE 3.9.3 NOTES

a Specific parameters of distance and direction sector from the centerline of the reactor and additional descriptions where pertinent, shall be provided for each and every sample location in Table 3.9.3 in a table and figure(s) in the ODCM. Deviations are permitted from the required sampling schedule if specimens are unobtainable due to hazardous conditions, seasonal unavailability, malfunction of automatic sampling equipment and other legitimate reasons. If specimens are unobtainable due to sampling equipment malfunction, every reasonable 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 Surveillance Report pursuant to Specification 6.7.C.3. 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 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. In lieu of a Licensee Event Report and pursuant to Specification 6.7.C.1, identify the cause of the unavailability of samples for that pathway and identify the new location(s) for obtaining replacement samples in the next Semiannual Radioactive Effluent Release Report and also include in the report a revised figure(s) and table for the ODCM reflecting the new location(s).

b 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 dosimeters. Film badges shall not be used as dosimeters for measuring direct radiation. The 40 stations is not an absolute number. 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.

c 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.

d Gamma isotopic analysis means the identification and quantification of gamma-emitting radionuclides that may be attributable to the effluents from the facility.

e 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.

f 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.

g Each meteorological sector shall have an established "inner" and an "outer" monitoring location based on ease of recovery (i.e., response time) and year-round accessibility.

h Sample collection will be performed weekly whenever the main plant stack effluent release rate of I-131, as determined by the sampling and analysis program of Table 4.8.2, is equal to or greater than 1×10^{-1} uCi/sec. Sample collection will revert back to semimonthly no sooner than at least two weeks after the plant stack effluent release rate of I-131 falls and remains below 1×10^{-1} uCi/sec.

TABLE 3.9.4

REPORTING LEVELS FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES[a]

Reporting Levels

Analysis	Water (pCi/l)	Airborne Particulate or Gases (pCi/m3)	Fish (pCi/Kg, wet)	Milk (pCi/l)	Vegetation (pCi/Kg, wet)	Sediment (pCi/Kg, dry)
H-3	2×10^4 [b]					
Mn-54	1×10^3		3×10^4			
Fe-59	4×10^2		1×10^4			
Co-58	1×10^3		3×10^4			
Co-60	3×10^2		1×10^4			3×10^3 [c]
Zn-65	3×10^2		2×10^4			
Zr-Nb-95	4×10^2					
I-131		0.9		3	1×10^2	
Cs-134	30	10	1×10^3	60	1×10^3	
Cs-137	50	20	2×10^3	70	2×10^3	
Ba-La-140	2×10^2			3×10^2		

(a) Reporting levels may be averaged over a calendar quarter. When more than one of the radionuclides in Table 3.9.4 are detected in the sampling medium, the unique reporting requirements are not exercised if the following condition holds:

$$\frac{\text{Concentration (1)}}{\text{Reporting Level (1)}} + \frac{\text{Concentration (2)}}{\text{Reporting Level (2)}} + \ldots \quad 1.0.$$

When radionuclides other than those in Table 3.9.4 are detected and are the result of plant effluents, the potential annual dose to a member of the public must be less than or equal to the calendar year limits of Specifications 3.8.B, 3.8.E, and 3.8.F.

(b) Reporting level for drinking water pathways. For nondrinking water pathways, a value of 3×10^4 pCi/l may be used.

(c) Reporting level for individual grab samples taken at North Storm Drain Outfall only.

TABLE 4.9.1

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Instrument	Instrument Check	Source Check	Instrument Calibration	Instrument Functional Test
1. Gross Radioactivity Monitors not Providing Automatic Termination of Release				
a. Liquid Radwaste Discharge Monitor (3)	Once each day*	Prior to each release, but no more than once each month	Once each 18 months (1)	Once each quarter (2)
b. Service Water Discharge Monitor (3)	Once each day	Once each month	Once each 18 months (1)	Once each quarter (2)
2. Flow Rate Measurement Devices				
a. Liquid Radwaste Discharge Flow Rate Monitor	Once each day*	Not Applicable	Not Applicable	Once each quarter*

* During releases via this pathway.

TABLE 4.9.1 NOTES

(1) The Instrument Calibration for radioactivity measurement instrumentation shall include the use of a known (traceable to National Bureau of Standards) liquid radioactive source positioned in a reproducible geometry with respect to the sensor. These standards shall permit calibrating the system over its normal operating range of energy and rate.

(2) The Instrument Functional Test shall also demonstrate the Control Room alarm annunciation occurs if any of the following conditions exists:

(a) Instrument indicate measured levels above the alarm setpoint.

(b) Circuit failure.

(c) Instrument indicates a downscale failure.

(d) Instrument controls not set in operate mode.

(3) The alarm setpoints of these channels shall be determined and adjusted in accordance with the methodology and parameters in the Off-Site Dose Calculation Manual (ODCM).

TABLE 4.9.2

GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Instrument	Instrument Check	Source Check	Instrument Calibration	Instrument Functional Test
1. Steam Jet Air Ejector (SJAE)				
a. Noble Gas Activity Monitor	Once each day**	Once each month	Once each 18 months (3)	Once each quarter (2)
2. Augmented Off-Gas System				
a. Noble Gas Activity Monitor	Once each day*	Once each month	Once each 18 months (3)	Once each quarter (1)
b. Flow Rate Monitor	Once each day*	Not Applicable	Once each 18 months	Not Applicable
c. Hydrogen Monitor	Once each day*	Not Applicable	Once each quarter (4)	Once each month
3. Plant Stack				
a. Noble Gas Activity Monitor	Once each day	Once each month	Once each 18 months (3)	Once each quarter (2)
b. Sampler Flow Integrator	Once each week	Not Applicable	Once each 18 months	Not Applicable
c. System Flow Rate Monitor	Once each day	Not Applicable	Not Applicable	Not Applicable

* During releases via this pathway.
** During operation of main condenser SJAE.

TABLE 4.9.2 NOTES

(1) The Instrument Functional Test shall also demonstrate that automatic isolation of this pathway and the Control Room alarm annunciation occurs if any of the following conditions exists:

(a) Instrument indicates measured levels above the alarm setpoint.

(b) Circuit failure.

(c) Instrument indicates a downscale failure.

(d) Instrument controls not set in operate mode.

(2) The instrument Functional Test shall also demonstrate that Control Room alarm annunciation occurs when any of the following conditions exist:

(a) Instrument indicates measured levels above the alarm setpoint.

(b) Circuit failure.

(c) Instrument indicates a downscale failure.

(d) Instrument controls are not set in operate mode.

(3) The Instrument Calibration for radioactivity measurement instrumentation shall include the use of a known (traceable to National Bureau of Standards) radioactive source positioned in a reproducible geometry with respect to the sensor. These standards should permit calibrating the system over its normal operating range of rate capabilities.

(4) The Instrument Calibration shall include the use of standard gas samples (high range and low range) containing suitable concentrations, hydrogen balance nitrogen, for the detection range of interest per Specification 3.8.J.1.

TABLE 4.9.3

DETECTION CAPABILITIES FOR ENVIRONMENTAL SAMPLE ANALYSIS[a][c][f]

Analysis[d]	Water (pCi/l)	Airborne Particulate or Gas (pCi/m3)	Fish (pCi/Kg, wet)	Milk (pCi/l)	Vegetation (pCi/Kg, wet)	Sediment (pCi/Kg, dry)
Gross beta	4	0.01				
H-3	3000					
Mn-54	15		130			
Fe-59	30		260			
Co-58, Co-60	15		130			
Zn-65	30		260			
Zr-Nb-95	15[b]					
I-131		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[b][e]			15[b][e]		

TABLE 4.9.3 NOTES

(a) See Footnote (a) of Table 4.8.1.

(b) Parent only.

(c) If the measured concentration minus the 5 sigma counting statistics is found to exceed the specified LLD, the sample does not have to be analyzed to meet the specified LLD.

(d) This list does not mean that only these nuclides are to be considered. Other peaks that are identifiable, together with those of the listed nuclides, shall also be analyzed and reported in the Annual Radiological Environmental Surveillance Report pursuant to Specification 6.7.C.3.

(e) The Ba-140 LLD and concentration can be determined by the analysis of its short-lived daughter product La-140 subsequent to an 8 day period following collection. The calculation shall be predicted on the normal ingrowth equations for a parent-daughter situation and the assumption that any unsupported La-140 in the sample would have decayed to an insignificant amount (at least 3.6 percent of its original value). The ingrowth equations will assume that the supported La-140 activity at the time of the collection is zero.

(f) Nuclides which are below the LLD for the analyses should not be reported as being present at the LLD, but as "not detected". For purposes of averaging, the LLD will be assumed to be zero.

BASES:

3.9 RADIOACTIVE EFFLUENT MONITORING SYSTEMS

A. Liquid Effluent Instrumentation

The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases of liquid effluents. The alarm setpoints for these instruments are to ensure that the alarm will occur prior to exceeding the limits of 10CFR Part 20.

Automatic isolation function is not provided on the liquid radwaste discharge line due to the infrequent nature of batch, discrete volume, liquid discharges (on the order of once per year or less), and the administrative controls provided to ensure that conservative discharge flow rates/dilution flows are set such that the probability of exceeding the 10CFR Part 20 concentration limits are low, and the potential off-site dose consequences are also low.

B. Gaseous Effluent Instrumentation

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The alarm/trip setpoints for these instruments are provided to ensure that the alarm/trip will occur prior to exceeding the limits of 10CFR Part 20. This instrumentation also includes provisions for monitoring (and controlling) the concentrations of potentially explosive gas mixtures in the waste gas holdup system.

C. Radiological Environmental Monitoring Program

The radiological monitoring program required by this specification provides measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides which lead to the highest potential radiation exposures of member(s) 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 modeling of the environmental exposure pathways.

Ten years of plant operation, including the years prior to the implementation of the Augmented Off-Gas System, have amply demonstrated via routine effluent and environmental reports that plant effluent measurements and modeling of environmental pathways are adequately conservative. In all cases, environmental sample results have been two to three orders of magnitude less than expected by the model employed, thereby representing small percentages of the ALARA and environmental reporting levels. This radiological environmental monitoring program has therefore been significantly modified as provided for by Regulatory Guides 4.3 (C.2.a) and 4.1 (C.2.b), Revision 1, April 1975. Specifically, the air particulate and radioiodine air sampling periods have been increased to semimonthly, based on plant effluent and environmental air sampling data for the previous ten years of operation. An I-131 release rate trigger value of 1×10^{-1} uCi/sec from the plant stack will require that air sample collection be increased to weekly. The

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1×10^{-1} uCi/sec I-131 value corresponds to the LLD air concentration of 0.07 pCi/m^3 at the maximum predicted air monitoring station, which exhibits a maximum quarterly X/Q value of 2×10^{-7} sec/m^3. A factor of 3.5 below the LLD value has also been included in the stack release rate value to account for meteorological fluctuations in X/Q. Due to the large local population of cows and the ready availability of milk samples, food product sampling has been eliminated from the program in lieu of milk sampling. Since milking cows in the area spend very little time on pasture, silage and grass sampling have been instituted as an indicator of radionuclide deposition.

The detection capabilities required by Table 4.9.3 are considered optimum for routine environmental measurements in industrial laboratories. It should be recognized that the LLD is defined as a before-the-fact limit representing the capability of a measurement system and not as an after-the-fact limit for a particular measurement. This does not preclude the calculation of an after-the-fact LLD for a particular measurement based upon the actual parameters for the sample in question.

D. Land Use Census

This specification is provided to ensure that changes in the use of areas at and beyond the site boundaries are identified and that modifications to the monitoring program are made if required by the results of this census. This census satisfies the requirements of Section IV.B.3 of Appendix I to 10 CFR Part 50. The requirement of a garden census has been eliminated along with the food product monitoring requirement due to the substantial and widespread occurrence of dairy farming in the surrounding area which dominates the food uptake pathway.

The addition of new sampling locations to Specification 3.9.C, based on the land use census, is limited to those locations which yield a calculated dose or dose commitment greater than 20 percent of the calculated dose or dose commitment at any location currently being sampled. This eliminates the unnecessary changing of the environmental radiation monitoring program for new locations which, within the accuracy of the calculation, contributes essentially the same to the dose or dose commitment as the location already sampled. The substitution of a new sampling point for one already sampled when the calculated difference in dose is less than 20 percent, would not be expected to result in a significant increase in the ability to detect plant effluent related nuclides.

E. Intercomparison Program

The requirement for participation in an intercomparison 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 a quality assurance program for environmental monitoring in order to demonstrate that the results are reasonably valid for the purposes of Section IV.B.2 of Appendix I to 10 CFR Part 50.

3.9 LIMITING CONDITIONS FOR OPERATION

3.9 RADIOACTIVE EFFLUENT MONITORING SYSTEMS

Applicability:

Applies to the monitoring systems or programs which perform a surveillance, protective or controlling function on the release of radioactive effluents from the plant and their identification in the environment.

Objective:

To assure the operability of the radioactive effluent monitoring systems and environmental programs.

Specifications:

A. Liquid Effluent Instrumentation

1. During periods of release through the monitored pathway, the radioactive liquid effluent monitoring instrumentation channel shall be operable in accordance with Table 3.9.1 with their alarm setpoints set to ensure that the limits of Specification 3.8.A.1 are not exceeded.

B. Gaseous Effluent Instrumentation

1. The gaseous process and effluent monitoring instrumentation channels shall be operable in accordance with Table 3.9.2 with their alarm/trip setpoints set to ensure that the limits of Specifications 3.8.E.1.a, 3.8.J.1, and 3.8.K.1 are not exceeded.

4.9 SURVEILLANCE REQUIREMENTS

4.9 RADIOACTIVE EFFLUENT MONITORING SYSTEMS

Applicability:

Applies to the required surveillance of the monitoring systems or programs which perform a surveillance, protective or controlling function on the release of radioactive effluents from the plant and their identification in the environment.

Objective:

To specify the type and frequency of surveillance to be applied to the radioactive effluent monitoring system and environmental programs.

Specifications:

A. Liquid Effluent Instrumentation

1. Each radioactive liquid effluent monitoring instrumentation channel shall be tested and calibrated as indicated in Table 4.9.1.

B. Gaseous Effluent Instrumentation

1. Each gaseous process or effluent monitoring instrumentation channel shall be tested and calibrated as indicated in Table 4.9.2.

3.9 LIMITING CONDITIONS FOR OPERATION

C. Radiological Environmental Monitoring Program

1. The radiological environmental monitoring program shall be conducted as specified in Table 3.9.3.

D. Land Use Census

1. A land use census shall be conducted to identify the location of the nearest milk animal and the nearest residence in each of the 16 meteorological sectors within a distance of five miles. The survey shall also identify the nearest milk animal (within 3 miles of the plant) to the point of predicted highest annual average D/Q value in each of the three major meteorological sectors due to elevated releases from the plant stack.

2. With a land use census identifying one or more locations which yield a calculated dose or dose commitment (via the same exposure pathway) at least 20 percent greater than at a location from which samples are currently being obtained in accordance with Specification 3.9.C.1, add the new location(s) to the radiological environmental monitoring program within 30 days if permission from the owner to collect samples can be obtained, and sufficient sample volume is available. The sampling location(s), excluding the control

4.9 SURVEILLANCE REQUIREMENTS

C. Radiological Environmental Monitoring Program

1. The radiological environmental monitoring samples shall be collected pursuant to Table 3.9.3 from the locations given in the ODCM and shall be analyzed pursuant to the requirements of Table 3.9.3 and the detection capabilities required by Table 4.9.3.

D. Land Use Census

1. The land use census shall be conducted at least once per year between the dates of June 1 and October 1 by either a door-to-door survey, aerial survey, or by consulting local agricultural authorities. The results of the land use census shall be included in the annual Radiological Environmental Surveillance Report pursuant to Specification 6.7.C.3.

3.9 LIMITING CONDITIONS FOR OPERATION

station location, having the lowest calculated dose or dose commitment (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.

E. Intercomparison Program

1. Analyses shall be performed on referenced radioactive materials supplied as part of an Intercomparison Program which has been approved by NRC.

4.9 SURVEILLANCE REQUIREMENTS

E. Intercomparison Program

1. A summary of the results of analyses performed as part of the above required Intercomparison Program shall be included in the Annual Radiological Environmental Surveillance Report. The identification of the NRC approved Intercomparison Program which is being participated in shall be stated in the ODCM.

TABLE 3.9.1

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

	Minimum Channels Operable	Notes
1. Gross Radioactivity Monitors not Providing Automatic Termination of Release		
a. Liquid Radwaste Discharge Monitor	1*	1,4,5
b. Service Water Discharge Monitor	1	2,4,5
2. Flow Rate Measurement Devices		
a. Liquid Radwaste Discharge Flow Rate Monitor	1*	3,4

* During releases via this pathway.

TABLE 3.9.1 NOTES

NOTE 1 - With the number of channels operable less than required by the minimum channels operable requirement, effluent releases may continue provided that prior to initiating a release:

a. At least two independent samples are analyzed in accordance with Specification 4.8.A.1, and

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

Otherwise, suspend release of radioactive effluents via this pathway.

NOTE 2 - With the number of channels operable less than required by the minimum channels operable requirement, effluent releases via this pathway may continue provided that, at least once per 24 hours, grab samples are collected and analyzed for gross radioactivity (beta or gamma) at a lower limit of detection of at least 10^{-7} microcurie/ml.

NOTE 3 - With the number of channels operable less than required by the minimum channels operable requirement, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 4 hours during actual releases. Pump performance curves may be used to estimate flow.

NOTE 4 - With the number of channels operable less than required by the minimum channels operable requirement, exert reasonable efforts to return the instrument(s) to operable status prior to the next release.

NOTE 5 - The alarm setpoints of these channels shall be determined and adjusted in accordance with the methodology and parameters in the Off-Site Dose Calculation Manual (ODCM). With a radioactive liquid effluent monitoring instrumentation channel alarm setpoint less conservative than a value which will ensure that the limits of e.8.A.1 are met during periods of release, immediately take action to suspend the release of radioactive liquid effluents monitored by the affected channel or declare the channel inoperable; or change the setpoint so it is acceptably conservative.

TABLE 3.9.2

GASEOUS EFFLUENT MONITORING INSTRUMENTATION

Instrument	Minimum Channels Operable	Notes
1. Steam Jet Air Ejector (SJAE)		
a. Noble Gas Activity Monitor	1	7, 8, 9
2. Augmented Off-Gas System		
a. Noble Gas Activity Monitor Between the Charcoal Bed System and the Plant Stack (Providing Alarm and Automatic Termination of Release)	1	2, 5, 6, 7
b. Flow Rate Monitor	1	1, 5, 6
c. Hydrogen Monitor	1	3, 5, 6
3. Plant Stack		
a. Noble Gas Activity Monitor	1	5, 7, 10
b. Iodine Sampler Cartridge	1	4, 5
c. Particulate Sampler Filter	1	4, 5
d. Sampler Flow Integrator	1	1, 5
e. Stack Flow Rate Monitor	1	1, 5

TABLE 3.9.2 NOTES

NOTE 1 - With the number of channels operable less than required by the minimum channels operable requirement, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 4 hours.

NOTE 2 - With the number of channels operable less than required by the minimum channels operable requirement, effluent releases via this pathway may continue for a period of up to 7 days provided that at least one of the stack monitoring systems is operable and off-gas system temperature and pressure are measured continuously.

NOTE 3 - With the number of channels operable less than required by the minimum channels operable requirement, operation of the AOG System may continue provided gas samples are collected at least once per 24 hours and analyzed within the following 4 hours, or an orderly transfer of the off-gas effluents from the operating recombiner to the standby recombiner shall be made.

NOTE 4 - With the number of channels operable less than required by the minimum channels operable requirement, effluent releases via the affected pathway may continue provided samples are continuously collected with auxiliary sampling equipment.

NOTE 5 - With the number of channels operable less than required by the minimum channels operable requirement, exert reasonable efforts to return the instrument(s) to operable status within 30 days.

NOTE 6 - During releases via this pathway.

NOTE 7 - The alarm/trip setpoints of these channels shall be determined and adjusted in accordance with the methodology and parameters in the Off-Site Dose Calculation Manual (ODCM). With a gaseous process or effluent monitoring instrumentation channel alarm/trip setpoint less conservative than a value which will ensure that the limits of 3.8.E.1.a and 3.8.K.1 are met, immediately take actions to suspend the release of radioactive gaseous effluents monitored by the affected channel, or declare the channel inoperable, or change the setpoint so it is acceptably conservative.

NOTE 8 - Minimum channels operable required only during operation of the Steam Jet Air Ejector.

NOTE 9 - With the number of channels operable less than required by the minimum channels operable requirement, gases from the SJAE may be released to the environment for up to 72 hours provided:

1. The AOG System is not bypassed; and

2. The AOG System noble gas activity monitor is operable.

NOTE 10 - With the number of channels operable less than required by the minimum channels operable requirement, effluent releases via this pathway may continue provided grab samples are taken at least once per 12 hours and these samples are analyzed for gross activity within 24 hours.

TABLE 3.9.3

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Sample Locations[a]	Sampling and Collection Frequency	Type and Frequency of Analysis
1. AIRBORNE			
a. Radioiodine and Particulates	Samples from 5 locations: 1 sample from up valley, within 4 miles of Site Boundary. (major wind direction) 1 sample from down valley, within 4 miles of Site Boundary. (major wind direction) 1 sample each from the vicinity of two nearby communities, within 10 miles of Site Boundary. 1 sample from a control location.	Continuous operation of sampler with sample collection semimonthly or more frequently as required by dust loading or plant effluent releases.[h]	Radioiodine canister: Analyze each sample for I-131. Particulate sampler: Gross beta radioactivity analysis on each sample following filter change.[c] Composite (by location) for gamma isotopic[d] at least once per quarter.

TABLE 3.9.3
(Cont'd)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Sample Locations[a]	Sampling and Collection Frequency	Type and Frequency of Analysis
2. DIRECT RADIATION[b]	40 routine monitoring stations as follows: 16 incident response stations (one in each meteorological sector) within a range of 0 to 4 km[8]; 16 incident response stations (one in each meteorological sector) within a range of 2 to 8 km[8]; the balance of the stations to be placed in special interest areas and control station areas.	Quarterly.	Gamma dose, at least once per quarter. Incident response TLDs in the outer monitoring locations, de-dose only quarterly unless gaseous release LCO was exceeded in period.

TABLE 3.9.3
(Cont'd)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Sample Locations[a]	Sampling and Collection Frequency	Type and Frequency of Analysis
3. WATERBORNE			
a. Surface[e]	1 sample upstream.	Monthly grab sample.	Gamma isotopic analysis[d] of each sample. Tritium analysis of composite sample at least once per quarter.
	1 sample downstream.	Composite sample collected over a period of one month[f].	
b. Ground	1 sample from within 8 km distance.	Quarterly.	Gamma isotopic[d] and tritium analyses of each sample.
	1 sample from a control location.	Quarterly.	
c. Sediment from Shoreline	1 sample from downstream area with existing or potential recreational value.	Semiannually.	Gamma isotopic analysis[d] of each sample.
	1 sample from north storm drain outfall.	As specified in the ODCM.	

TABLE 3.9.3
(Cont'd)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Sample Locations[a]	Sampling and Collection Frequency	Type and Frequency of Analysis
4. INGESTION			
a. Milk	Samples from milking animals in 3 locations within 5 km distance having the highest dose potential. If there are less than 3 primary locations available then 1 or more secondary sample from milking animals in each of 3 areas between 5 and 8 km distance where doses are calculated to be greater than 1 mrem per year.	Semimonthly if milking animals are identified on pasture; at least once per month at other times.	Gamma isotopic[d] and I-131 analysis of each sample.
	1 sample from milking animals in a control location.		
b. Fish	1 sample of two recreationally important species in vicinity of plant discharge area.	Semiannually.	Gamma isotopic analysis[d] on edible portions.
	1 sample (preferably of same species) in areas not influenced by plant discharge.		
c. Vegetation	1 grass sample at each air sampling station.	Quarterly when available.	Gamma isotopic analysis[d] of each sample.
	1 silage sample at each milk sampling station (as available).	At time of harvest.	Gamma isotopic analysis[d] of each sample.

TABLE 3.9.3 NOTES

a Specific parameters of distance and direction sector from the centerline of the reactor and additional descriptions where pertinent, shall be provided for each and every sample location in Table 3.9.3 in a table and figure(s) in the ODCM. Deviations are permitted from the required sampling schedule if specimens are unobtainable due to hazardous conditions, seasonal unavailability, malfunction of automatic sampling equipment and other legitimate reasons. If specimens are unobtainable due to sampling equipment malfunction, every reasonable 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 Surveillance Report pursuant to Specification 6.7.C.3. 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 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. In lieu of a Licensee Event Report and pursuant to Specification 6.7.C.1, identify the cause of the unavailability of samples for that pathway and identify the new location(s) for obtaining replacement samples in the next Semiannual Radioactive Effluent Release Report and also include in the report a revised figure(s) and table for the ODCM reflecting the new location(s).

b 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 dosimeters. Film badges shall not be used as dosimeters for measuring direct radiation. The 40 stations is not an absolute number. 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.

c 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.

d Gamma isotopic analysis means the identification and quantification of gamma-emitting radionuclides that may be attributable to the effluents from the facility.

e 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.

f 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.

g Each meteorological sector shall have an established "inner" and an "outer" monitoring location based on ease of recovery (i.e., response time) and year-round accessibility.

h Sample collection will be performed weekly whenever the main plant stack effluent release rate of I-131, as determined by the sampling and analysis program of Table 4.8.2, is equal to or greater than 1×10^{-1} uCi/sec. Sample collection will revert back to semimonthly no sooner than at least two weeks after the plant stack effluent release rate of I-131 falls and remains below 1×10^{-1} uCi/sec.

TABLE 3.9.4

REPORTING LEVELS FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES[(a)]

Reporting Levels

Analysis	Water (pCi/l)	Airborne Particulate or Gases (pCi/m3)	Fish (pCi/Kg, wet)	Milk (pCi/l)	Vegetation (pCi/Kg, wet)	Sediment (pCi/Kg, dry)
H-3	2×10^4 [(b)]					
Mn-54	1×10^3		3×10^4			
Fe-59	4×10^2		1×10^4			
Co-58	1×10^3		3×10^4			
Co-60	3×10^2		1×10^4			3×10^3 [(c)]
Zn-65	3×10^2		2×10^4			
Zr-Nb-95	4×10^2					
I-131		0.9		3	1×10^2	
Cs-134	30	10	1×10^3	60	1×10^3	
Cs-137	50	20	2×10^3	70	2×10^3	
Ba-La-140	2×10^2			3×10^2		

(a) Reporting levels may be averaged over a calendar quarter. When more than one of the radionuclides in Table 3.9.4 are detected in the sampling medium, the unique reporting requirements are not exercised if the following condition holds:

$$\frac{\text{Concentration (1)}}{\text{Reporting Level (1)}} + \frac{\text{Concentration (2)}}{\text{Reporting Level (2)}} + \ldots \quad 1.0.$$

When radionuclides other than those in Table 3.9.4 are detected and are the result of plant effluents, the potential annual dose to a member of the public must be less than or equal to the calendar year limits of Specifications 3.8.B, 3.8.E, and 3.8.F.

(b) Reporting level for drinking water pathways. For nondrinking water pathways, a value of 3×10^4 pCi/l may be used.

(c) Reporting level for individual grab samples taken at North Storm Drain Outfall only.

TABLE 4.9.1

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Instrument	Instrument Check	Source Check	Instrument Calibration	Instrument Functional Test
1. Gross Radioactivity Monitors not Providing Automatic Termination of Release				
a. Liquid Radwaste Discharge Monitor (3)	Once each day*	Prior to each release, but no more than once each month	Once each 18 months (1)	Once each quarter (2)
b. Service Water Discharge Monitor (3)	Once each day	Once each month	Once each 18 months (1)	Once each quarter (2)
2. Flow Rate Measurement Devices				
a. Liquid Radwaste Discharge Flow Rate Monitor	Once each day*	Not Applicable	Not Applicable	Once each quarter*

* During releases via this pathway.

TABLE 4.9.1 NOTES

(1) The Instrument Calibration for radioactivity measurement instrumentation shall include the use of a known (traceable to National Bureau of Standards) liquid radioactive source positioned in a reproducible geometry with respect to the sensor. These standards shall permit calibrating the system over its normal operating range of energy and rate.

(2) The Instrument Functional Test shall also demonstrate the Control Room alarm annunciation occurs if any of the following conditions exists:

(a) Instrument indicate measured levels above the alarm setpoint.

(b) Circuit failure.

(c) Instrument indicates a downscale failure.

(d) Instrument controls not set in operate mode.

(3) The alarm setpoints of these channels shall be determined and adjusted in accordance with the methodology and parameters in the Off-Site Dose Calculation Manual (ODCM).

TABLE 4.9.2

GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Instrument	Instrument Check	Source Check	Instrument Calibration	Instrument Functional Test
1. Steam Jet Air Ejector (SJAE)				
a. Noble Gas Activity Monitor	Once each day**	Once each month	Once each 18 months (3)	Once each quarter (2)
2. Augmented Off-Gas System				
a. Noble Gas Activity Monitor	Once each day*	Once each month	Once each 18 months (3)	Once each quarter (1)
b. Flow Rate Monitor	Once each day*	Not Applicable	Once each 18 months	Not Applicable
c. Hydrogen Monitor	Once each day*	Not Applicable	Once each quarter (4)	Once each month
3. Plant Stack				
a. Noble Gas Activity Monitor	Once each day	Once each month	Once each 18 months (3)	Once each quarter (2)
b. Sampler Flow Integrator	Once each week	Not Applicable	Once each 18 months	Not Applicable
c. System Flow Rate Monitor	Once each day	Not Applicable	Not Applicable	Not Applicable

* During releases via this pathway.
** During operation of main condenser SJAE.

TABLE 4.9.2 NOTES

(1) The Instrument Functional Test shall also demonstrate that automatic isolation of this pathway and the Control Room alarm annunciation occurs if any of the following conditions exists:

(a) Instrument indicates measured levels above the alarm setpoint.

(b) Circuit failure.

(c) Instrument indicates a downscale failure.

(d) Instrument controls not set in operate mode.

(2) The instrument Functional Test shall also demonstrate that Control Room alarm annunciation occurs when any of the following conditions exist.

(a) Instrument indicates measured levels above the alarm setpoint.

(b) Circuit failure.

(c) Instrument indicates a downscale failure.

(d) Instrument controls are not set in operate mode.

(3) The Instrument Calibration for radioactivity measurement instrumentation shall include the use of a known (traceable to National Bureau of Standards) radioactive source positioned in a reproducible geometry with respect to the sensor. These standards should permit calibrating the system over its normal operating range of rate capabilities.

(4) The Instrument Calibration shall include the use of standard gas samples (high range and low range) containing suitable concentrations, hydrogen balance nitrogen, for the detection range of interest per Specification 3.8.J.1.

TABLE 4.9.3

DETECTION CAPABILITIES FOR ENVIRONMENTAL SAMPLE ANALYSIS(a)(c)(f)

Analysis(d)	Water (pCi/l)	Airborne Particulate or Gas (pCi/m3)	Fish (pCi/Kg, wet)	Milk (pCi/l)	Vegetation (pCi/Kg, wet)	Sediment (pCi/Kg, dry)
Gross beta	4	0.01				
H-3	3000					
Mn-54	15		130			
Fe-59	30		260			
Co-58, Co-60	15		130			
Zn-65	30		260			
Zr-Nb-95	15(b)					
I-131		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(b)(e)			15(b)(e)		

TABLE 4.9.3 NOTES

(a) See Footnote (a) of Table 4.8.1.

(b) Parent only.

(c) If the measured concentration minus the 5 sigma counting statistics is found to exceed the specified LLD, the sample does not have to be analyzed to meet the specified LLD.

(d) This list does not mean that only these nuclides are to be considered. Other peaks that are identifiable, together with those of the listed nuclides, shall also be analyzed and reported in the Annual Radiological Environmental Surveillance Report pursuant to Specification 6.7.C.3.

(e) The Ba-140 LLD and concentration can be determined by the analysis of its short-lived daughter product La-140 subsequent to an 8 day period following collection. The calculation shall be predicted on the normal ingrowth equations for a parent-daughter situation and the assumption that any unsupported La-140 in the sample would have decayed to an insignificant amount (at least 3.6 percent of its original value). The ingrowth equations will assume that the supported La-140 activity at the time of the collection is zero.

(f) Nuclides which are below the LLD for the analyses should not be reported as being present at the LLD, but as "not detected". For purposes of averaging, the LLD will be assumed to be zero.

BASES:

3.9 RADIOACTIVE EFFLUENT MONITORING SYSTEMS

A. Liquid Effluent Instrumentation

The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases of liquid effluents. The alarm setpoints for these instruments are to ensure that the alarm will occur prior to exceeding the limits of 10CFR Part 20.

Automatic isolation function is not provided on the liquid radwaste discharge line due to the infrequent nature of batch, discrete volume, liquid discharges (on the order of once per year or less), and the administrative controls provided to ensure that conservative discharge flow rates/dilution flows are set such that the probability of exceeding the 10CFR Part 20 concentration limits are low, and the potential off-site dose consequences are also low.

B. Gaseous Effluent Instrumentation

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The alarm/trip setpoints for these instruments are provided to ensure that the alarm/trip will occur prior to exceeding the limits of 10CFR Part 20. This instrumentation also includes provisions for monitoring (and controlling) the concentrations of potentially explosive gas mixtures in the waste gas holdup system.

C. Radiological Environmental Monitoring Program

The radiological monitoring program required by this specification provides measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides which lead to the highest potential radiation exposures of member(s) 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 modeling of the environmental exposure pathways.

Ten years of plant operation, including the years prior to the implementation of the Augmented Off-Gas System, have amply demonstrated via routine effluent and environmental reports that plant effluent measurements and modeling of environmental pathways are adequately conservative. In all cases, environmental sample results have been two to three orders of magnitude less than expected by the model employed, thereby representing small percentages of the ALARA and environmental reporting levels. This radiological environmental monitoring program has therefore been significantly modified as provided for by Regulatory Guides 4.3 (C.2.a) and 4.1 (C.2.b), Revision 1, April 1975. Specifically, the air particulate and radioiodine air sampling periods have been increased to semimonthly, based on plant effluent and environmental air sampling data for the previous ten years of operation. An I-131 release rate trigger value of 1×10^{-1} uCi/sec from the plant stack will require that air sample collection be increased to weekly. The

BASES: 3.9 (Cont'd)

1×10^{-1} uCi/sec I-131 value corresponds to the LLD air concentration of 0.07 pCi/m^3 at the maximum predicted air monitoring station, which exhibits a maximum quarterly X/Q value of 2×10^{-7} sec/m^3. A factor of 3.5 below the LLD value has also been included in the stack release rate value to account for meteorological fluctuations in X/Q. Due to the large local population of cows and the ready availability of milk samples, food product sampling has been eliminated from the program in lieu of milk sampling. Since milking cows in the area spend very little time on pasture, silage and grass sampling have been instituted as an indicator of radionuclide deposition.

The detection capabilities required by Table 4.9.3 are considered optimum for routine environmental measurements in industrial laboratories. It should be recognized that the LLD is defined as a before-the-fact limit representing the capability of a measurement system <u>and not</u> as an after-the-fact limit for a particular measurement. This does not preclude the calculation of an after-the-fact LLD for a particular measurement based upon the actual parameters for the sample in question.

D. Land Use Census

This specification is provided to ensure that changes in the use of areas at and beyond the site boundaries are identified and that modifications to the monitoring program are made if required by the results of this census. This census satisfies the requirements of Section IV.B.3 of Appendix I to 10 CFR Part 50. The requirement of a garden census has been eliminated along with the food product monitoring requirement due to the substantial and widespread occurrence of dairy farming in the surrounding area which dominates the food uptake pathway.

The addition of new sampling locations to Specification 3.9.C, based on the land use census, is limited to those locations which yield a calculated dose or dose commitment greater than 20 percent of the calculated dose or dose commitment at any location currently being sampled. This eliminates the unnecessary changing of the environmental radiation monitoring program for new locations which, within the accuracy of the calculation, contributes essentially the same to the dose or dose commitment as the location already sampled. The substitution of a new sampling point for one already sampled when the calculated difference in dose is less than 20 percent, would not be expected to result in a significant increase in the ability to detect plant effluent related nuclides.

E. Intercomparison Program

The requirement for participation in an intercomparison 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 a quality assurance program for environmental monitoring in order to demonstrate that the results are reasonably valid for the purposes of Section IV.B.2 of Appendix I to 10 CFR Part 50.

3.10 LIMITING CONDITIONS FOR OPERATION

3.10 AUXILIARY ELECTRICAL POWER SYSTEMS

Applicability:

Applies to the auxiliary electrical power systems.

Objective:

To assure an adequate supply of electrical power for operation of those systems required for reactor safety.

Specification:

A. Normal Operation

The reactor shall not be made critical unless all of the following conditions are satisfied.

1. Diesel Generators

Both emergency diesel generators shall be operable and capable of starting and reaching rated voltage and frequency in not more than 13 seconds.

4.10 SURVEILLANCE REQUIREMENTS

4.10 AUXILIARY ELECTRICAL POWER SYSTEMS

Applicability:

Applies to the periodic testing requirements of the auxiliary electrical power systems.

Objective:

To verify the operability of the auxiliary electrical power systems.

Specification:

A. Normal Operation

1. Diesel Generators

a. Each diesel generator shall be started and loaded once a month to demonstrate operational readiness. The test shall continue until the diesel engine and the generator are at equilibrium temperature at expected maximum emergency loading not to exceed the continuous rating. During this test, the diesel starting time to reach rated voltage and frequency shall be logged and the air compressor shall be checked for operation and its ability to recharge air receivers. The diesel fuel oil transfer pumps shall be tested in accordance with Specification 4.6.E.

3.10 LIMITING CONDITIONS FOR OPERATION

2. Battery Systems

The following battery systems shall be operable:

a. The four Neutron Monitoring and Process Radiation Batteries, associated chargers, and 24 VDC Distribution Panels.

b. The two main station battery systems consisting of:

1. Battery A1, Battery Charger A or Spare Charger AB and Bus DC-1.

2. Battery B1, Battery Charger B or Spare Charger AB and Bus DC-2.

4.10 SURVEILLANCE REQUIREMENTS

b. The undervoltage automatic starting circuit of each diesel generator shall be tested once a month.

c. Once per operating cycle, the actual conditions under which the diesel generators are required to start automatically will be simulated and a test conducted to demonstrate that they will start within 13 seconds and accept the emergency load and start each load within the specified starting time. The results shall be logged.

2. Battery Systems

a. Every week the specific gravity, temperature, level, and voltage of the pilot cell and overall battery voltage shall be measured and logged.

b. Every three months the voltage, temperature, level, and specific gravity of each cell, and overall battery voltage shall be measured and logged.

3.10 LIMITING CONDITIONS FOR OPERATION

c. Two Switchyard Batteries each with one associated charger and its associated DC distribution panel.

d. Both ECCS Instrumentation batteries, associated chargers, and distribution panels.

e. The Alternate Shutdown AS-2 battery, one of the two associated chargers, and DC Distribution panel DC-2AS.

f. Both UPS batteries, associated Uninterruptible Power Supplies and MCC 89A and B.

4.10 SURVEILLANCE REQUIREMENTS

c. Once per operating cycle each ECCS battery, Alternate Shutdown AS-2 battery, and Main Station battery shall be subjected to a Service (Load Profile) discharge test. The specific gravity and voltage of each cell shall be measured after the recharge at the end of the discharge test and logged.

d. Once every five years, each ECCS, UPS, AS-2, and Main Station Battery shall be subjected to a Performance (capacity) Discharge Test. This test will be performed in lieu of the Service Test requirements of 4.10.A.2.c above.

e. Each 480 V Uninterruptible Power System shall be checked daily.

f. 480 V Motor Control Centers 89A and 89B shall be checked daily.

g. Once per operating cycle, the actual conditions under which the 480 V Uninterruptible Power Systems are required will be simulated and a test conducted to demonstrate equipment performance.

3.10 LIMITING CONDITIONS FOR OPERATION

3. Emergency Buses

The emergency 4160 volt Buses 3 and 4, and 480 volt Buses 8 and 9 shall be energized and operable.

4. Off-Site Power

a. At least one off-site transmission line and at least one start-up transformer in service.

b. One of the following additional sources of delayed access power:

The main stepup transformer and unit auxiliary transformer available and capable of supplying power to the emergency 4160 volt buses or,

The 4160 volt tie line to Vernon Hydro-Electric station capable of supplying power to either of the two emergency 4160 volt buses.

5. Reactor Protection System Power Protection

Two RPS power protection panels for each inservice RPS MG set or alternate power source shall be operable.

4.10 SURVEILLANCE REQUIREMENTS

3. Emergency Buses

The emergency 4160 volt buses and 480 volt buses shall be checked daily.

4. Off-Site Power

The status of the off-site power sources shall be checked daily.

5. Reactor Protection System Power Protection

Once per operating cycle, the operability of each overvoltage, undervoltage, and underfrequency protective device shall be demonstrated by the performance of an instrument channel calibration test. Settings shall be verified to be in accordance with Table 4.10.1.

3.10 LIMITING CONDITIONS FOR OPERATION

B. Operation With Inoperable Components

Whenever the reactor is in Run Mode or Startup Mode with the reactor not in the Cold Condition, the requirements of 3.10.A shall be met except:

1. Diesel Generators

From and after the date that one of the diesel generators or its associated buses are made or found to be inoperable for any reason and the remaining diesel generator is operable, the requirements of Specification 3.5.H.1 shall be satisfied.

2. Batteries

a. From and after the date that ventilation is lost in the Battery Room portable ventilation equipment shall be provided.

b. From and after the date that one of the two 125 volt Station Battery Systems is made or found to be inoperable for any reasons, continued reactor operation is permissible only during the succeeding three days provided Specification 3.5.H is met unless such Battery System is sooner made operable.

4.10 SURVEILLANCE REQUIREMENTS

B. Operation With Inoperable Components

1. Diesel Generator

When one of the diesel generators is made or found to be inoperable, the requirements of Specification 4.5.H.1 shall be satisfied.

2. Batteries

Samples of the Battery Room atmosphere shall be taken daily for hydrogen concentration determination.

3.10 LIMITING CONDITIONS FOR OPERATION	4.10 SURVEILLANCE REQUIREMENTS
c. From and after the date that one of the two 24 volt ECCS Instrumentation Battery Systems is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding three days unless such Battery System is sooner made operable.	
d. From and after the date that the AS-2 125 Volt battery system is made or found to be inoperable for any reason, continued reactor operation is permissible provided Diesel Generator DG-1-1A control power is transferred to Station Battery B1 and a fire watch is established to inspect the cable vault a minimum of every two hours.	
e. From and after the date that one of the two 24 Volt Neutron Monitoring and Process Radiation Monitoring battery systems is found or made to be inoperable for any reason, continued reactor operation is permissible providing the minimum channel requirements of Sections 3.1 and 3.2 for the Neutron Monitoring and Process Radiation Monitoring systems are met.	

3.10 LIMITING CONDITIONS FOR OPERATION

f. From and after the date that one of the two 125 volt Switchyard battery systems is found or made to be inoperable for any reason, continued reactor operation is permissible provided that the other 125 volt Switchyard battery system is operable.

3. Off-Site Power

a. From and after the date that both startup transformers and one diesel generator or associated buses are made or found to be inoperable for any reason, reactor operation may continue provided the requirements of Specification 3.5.H.1 are satisfied.

b. From and after the date that both delayed access off-site power sources become unavailable, reactor operation may continue for seven days provided both emergency diesel generators, associated buses, and all Low Pressure Core and Containment Cooling Systems are operable.

4.10 SURVEILLANCE REQUIREMENTS

3. Off-Site Power

a. When one of the diesel generators or associated buses is made or found to be inoperable, the requirements of Specification 4.5.H.1 shall be satisfied.

b. When both delayed access off-site power sources are unavailable, both diesel generators and associated buses shall have been or shall be demonstrated to be operable within 24 hours.

3.10 LIMITING CONDITIONS FOR OPERATION

4. 480 V Uninterruptible Power Systems

From and after the date that one Uninterruptible Power System or its associated Motor Control Center are made or found to be inoperable for any reason, the requirements of Specification 3.5.A.4 shall be satisfied.

5. RPS Power Protection

From and after the date that one of the two redundant RPS power protection panels on an in-service RPS MG set or alternate power supply is made or found to be inoperable, the associated RPS MG set or alternate supply will be taken out of service until the panel is restored to operable status.

C. Diesel Fuel

There shall be a minimum of 25,000 usable gallons of diesel fuel in the diesel fuel oil storage tank.

4.10 SURVEILLANCE REQUIREMENTS

4. 480 V Uninterruptible Power Systems

When it is determined that one Uninterruptible Power System or its associated Motor Control Center is inoperable, the requirements of Specification 4.5.A.4 shall be satisfied.

C. Diesel Fuel

1. The quantity of diesel generator fuel shall be logged weekly and after each operation of the unit.

2. Once a month a sample of diesel fuel shall be taken and checked for quality. The quality shall be within the applicable limits specified on Table I of ASTM D975-68 and logged.

TABLE 4.10.1

REACTOR PROTECTION SYSTEM POWER PROTECTION

Parameter	Setpoints for Panels A1, A2, B1, B2, C1, and C2
Overvoltage	$\leq$ 125.5 volts
Overvoltage Time Delay	$\leq$ 0.35 seconds
Undervoltage	$\geq$ 111 volts
Undervoltage Time Delay	$\leq$ 0.35 seconds
Underfrequency	$\geq$ 56.5 Hz
Underfrequency Time Delay	$\leq$ 0.35 seconds

BASES:

3.10 AUXILIARY ELECTRIC POWER SYSTEMS

A. The objective of this Specification is to assure that adequate power will be available to operate the emergency safeguards equipment. Adequate power can be provided by any one of the following sources: either of the startup transformers, backfeed through the main transformer, the 4160 volt line from the Vernon Hydroelectric Station or either of the two diesel generators. The backfeed through the main transformer and 4160 volt Vernon line are both delayed access off-site power sources. Backfeeding through the main transformer can be accomplished by disconnecting the main generator from the main transformer and energizing the auxiliary transformer from the 345 kV switchyard through the main transformer. The time required to perform this disconnection is approximately six hours. The 4160 volt line from the Vernon Hydroelectric Station can be connected to either of the two emergency buses within seconds by simple manual switching operation in the Main Control Room.

Two 480 V Uninterruptible Power Systems supply power to the LPCIS valves via designated Motor Control Centers. The 480 V Uninterruptible Power Systems are redundant and independent of any on-site power sources.

This Specification assures that at least two off-site and two on-site power sources, and both 480 V Uninterruptible Power Systems will be available before the reactor is taken beyond "just critical" testing. In addition to assuring power source availability, all of the associated switchgear must be operable as specified to assure that the emergency cooling equipment can be operated, if required, from the power sources.

Station service power is supplied to the station through either the unit auxiliary transformer or the startup transformers. In order to start up the station, at least one startup transformer is required to supply the station auxiliary load. After the unit is synchronized to the system, the unit auxiliary transformer carries the station auxiliary load, except for the station cooling tower loads which are always supplied by one of the startup transformers. The station cooling tower loads are not required to perform an engineered safety feature function in the event of an accident; therefore, an alternate source of power is not essential. Normally one startup transformer supplies 4160 volt Buses 1 and 3, and the other supplies Buses 2 and 4; however, the two startup transformers are designed with adequate capacity such that, should one become or be made inoperable, temporary connections can be made to supply the total station load (less the cooling towers) from the other startup transformer.

A battery charger is supplied for each battery. In addition, the two 125 volt station batteries have a spare charger available. Since one spare 125 volt station charger is available, one station battery charger can be allowed out of service for maintenance and repairs.

Power for the Reactor Protection System is supplied by 120 V ac motor generators with an alternate supply from MCC-8B. Two redundant, Class 1E, seismically qualified power protection panels are connected in series with each ac power source. These panels provide overvoltage, undervoltage, and underfrequency protection for the system. Setpoints are chosen to be consistent with the input power requirements of the equipment connected to the bus.

BASES: 3.10 (Cont'd)

B. Adequate power is available to operate the emergency safeguards equipment from either startup transformer or for minimum engineered safety features from either of the emergency diesel generators. Therefore, reactor operation is permitted for up to seven days with both delayed-access off-site power sources lost.

Each of the diesel generator units is capable of supplying 100 percent of the minimum emergency loads required under postulated design basis accident conditions. Each unit is physically and electrically independent of the other and of any off-site power source. Therefore, one diesel generator can be allowed out of service for a period of seven days without jeopardizing the safety of the station.

In the event that both startup transformers are lost, adequate power is available to operate the emergency safeguards equipment from either of the emergency diesel generators or from either of the delayed-access off-site power sources. Also, in the event that both emergency diesel generators are lost, adequate power is available immediately to operate the emergency safeguards equipment from at least one of the startup transformers or from either of the delayed-access off-site power sources within six hours. The plant is designed to accept one hundred percent load rejection without adverse effects to the plant or the transmission system. Network stability analysis studies indicate that the loss of the Vermont Yankee unit will not cause instability and consequent tripping of the connecting 345 kV and 115 kV lines. The Vernon feed is an independent source. Thus, the availability of the delayed-access off-site power sources is assured in the event of a turbine trip. Therefore, reactor operation is permitted with the startup transformers out of service and with one diesel generator out of service provided the NRC is notified immediately of the event and restoration plans.

Either of the two station batteries has enough capacity to energize the vital buses and supply d-c power to the other emergency equipment for 8 hours without being recharged. In addition, two 24 volt ECCS Instrumentation batteries supply power to instruments that provide automatic initiation of the ECCS and some reactor pressure and level indication in the Control Room.

Due to the high reliability of battery systems, one of the two batteries may be out of service for up to three days. This minimizes the probability of unwarranted shutdown by providing adequate time for reasonable repairs. A station battery, ECCS Instrumentation battery, or an Uninterruptible Power System battery is considered inoperable if more than one cell is out of service. A cell will be considered out of service if its float voltage is below 2.13 volts and the specific gravity is below 1.190 at 77°F.

The Battery Room is ventilated to prevent accumulation of hydrogen gas. With a complete loss of the ventilation system, the accumulation of hydrogen would not exceed 4 percent concentration in 16 days. Therefore, on loss of Battery Room ventilation, the use of portable ventilation equipment and daily sampling provide assurance that potentially hazardous quantities of hydrogen gas will not accumulate.

C. The minimum diesel fuel supply of 25,000 gallons will supply one diesel generator for a minimum of seven days of operation satisfying the load requirements for the operation of the safeguards equipment. Additional fuel can be obtained and delivered to the site from nearby sources within the seven-day period.

BASES:

4.10 AUXILIARY ELECTRICAL POWER SYSTEMS

A. The monthly tests of the diesel generators are conducted to check for equipment failures and deterioration. The test of the undervoltage automatic starting circuits will prove that each diesel will receive a start signal if a loss of voltage should occur on its emergency bus. The loading of each diesel generator is conducted to demonstrate proper operation at less than the continuous rating and at equilibrium operating conditions. Generator experience at other generator stations indicates that the testing frequency is adequate to assure a high reliability of operation should the system be required.

Both diesel generators have air compressors and air receivers tanks for starting. It is expected that the air compressors will run only infrequently. During the monthly check of the units, each receiver will be drawn down below the point at which the compressor automatically starts to check operation and the ability of the compressors to recharge the receivers.

Following the tests of the units and at least weekly, the fuel volume remaining will be checked. At the end of the monthly load test of the diesel generators, the fuel oil transfer pump will be operated to refill the day tank. The day tank level indicator and alarm switches will be checked at this time. Fuel oil transfer pump operability testing is in accordance with Specification 4.6.E.

The test of the diesels and Uninterruptible Power Systems during each refueling interval will be more comprehensive in that it will functionally test the system; i.e., it will check starting and closure of breakers and sequencing of loads. The units will be started by simulation of a loss of coolant accident. In addition, a loss of normal power condition will be imposed to simulate a loss of off-site power. The timing sequence will be checked to assure proper loading in the time required. Periodic tests between refueling intervals check the capability of the diesels to start in the required time and to deliver the expected emergency load requirements. Periodic testing of the various components plus a functional test at a refueling interval are sufficient to maintain adequate reliability.

B. Although the Main Station, ECCS, AS-2, and UPS batteries will deteriorate with time, utility experience indicates there is almost no possibility of precipitous failure. The type of surveillance described in this specification is that which has been demonstrated over the years to provide an indication of a cell becoming irregular or unserviceable long before it becomes a failure.

The performance discharge test provides adequate indication and assurance that the batteries have the specified ampere hour capacity. The rate of discharge during this test shall be in accordance with the manufacturer's discharge characteristic curves for the associated batteries. The results of these tests will be logged and compared with the manufacturer's recommendations of acceptability.

The service discharge test provides a test of the batteries ability to satisfy the design requirements (battery duty cycle) of the associated dc system. This test will be performed using simulated or actual loads at the rates and for the durations specified in the design load profile.

BASES: 4.10 (Cont'd)

C. Logging the diesel fuel supply weekly and after each operation assures that the minimum fuel supply requirements will be maintained. During the monthly test for quality of the diesel fuel oil, a viscosity test and water and sediment test will be performed as described in ASTM D975-68. The quality of the diesel fuel oil will be acceptable if the results of the tests are within the limiting requirements for diesel fuel oils shown on Table 1 of ASTM D975-68.

3.11 LIMITING CONDITIONS FOR OPERATION	4.11 SURVEILLANCE REQUIREMENTS

3.11 REACTOR FUEL ASSEMBLIES

Applicability:

The Limiting Conditions for Operation associated with the fuel rods apply to these parameters which monitor the fuel rod operating conditions.

Objective:

The Objective of the Limiting Conditions for Operation is to assure the performance of the fuel rods.

Specifications:

A. Average Planar Linear Heat Generation Rate (APLHGR)

During steady state power operation, the APLHGR for each type of fuel as a function of average planar exposure shall not exceed the limiting values provided in the Core Operating Limits Report. For single recirculation loop operation, the limiting values shall be the values provided in the Core Operating Limits Report listed under the heading "Single Loop Operation." If at any time during steady-state operation it is determined by normal surveillance that the limiting value for APLHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the APLHGR is not returned to within prescribed limits within two (2) hours, the reactor shall be brought to the shutdown conditions within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

4.11 REACTOR FUEL ASSEMBLIES

Applicability:

The Surveillance Requirements apply to the parameters which monitor the fuel rod operating conditions.

Objective:

The Objective of the Surveillance Requirements is to specify the type and frequency of surveillance to be applied to the fuel rods.

Specifications:

A. Average Planar Linear Heat Generation Rate (APLHGR)

The APLHGR for each type of fuel as a function of average planar exposure shall be determined daily during reactor operation at $\geq$25% rated thermal power.

3.11 LIMITING CONDITIONS FOR OPERATION

B. Linear Heat Generation Rate (LHGR)

During steady state power operation, the linear heat generation rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed the maximum allowable LHGR provided in the Core Operating Limits Report.

If at any time during steady state operation it is determined by normal surveillance that the limiting value for LHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the LHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

4.11 SURVEILLANCE REQUIREMENTS

B. Linear Heat Generation Rate (LHGR)

The LHGR as a function of core height shall be checked daily during reactor operation at ≥25% rated thermal power.

3.11 LIMITING CONDITIONS FOR OPERATION

C. Minimum Critical Power Ratio (MCPR)

1. During steady state power operation the MCPR operating value shall be equal to or greater than the MCPR limits provided in the Core Operating Limits Report. For single recirculation loop operation, the MCPR Limits at rated flow are also provided in the Core Operating Limits Report. For core flows other than rated, the Operating MCPR Limit shall be the above value multiplied by K_f where K_f is provided in the Core Operating Limits Report. If at any time during steady-state operation it is determined by normal surveillance that the limiting value for MCPR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the steady-state MCPR is not returned to within the prescribed limits within two (2) hours, the reactor power shall be brought to shutdown condition, within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

4.11 SURVEILLANCE REQUIREMENTS

C. Minimum Critical Power Ratio (MCPR)

MCPR shall be determined daily during reactor power operation at $\geq$25% rated thermal power and following any change in power level or distribution that would cause operation with a limiting control rod pattern as described in the bases for Specification 3.3.B.6.

BASES:

3.11 FUEL RODS

A. Average Planar Linear Heat Generation Rule (APLHGR)

Refer to the appropriate topical reports listed in Specification 6.7.A.4 for analyses methods.

(Note: All exposure increments in this Technical Specification section are expressed in terms of megawatt-days per short ton.)

The MAPLHGR reduction factor of 0.83 for single recirculation loop operation is based on the assumption that the coastdown flow from the unbroken recirculation loop would not be available during a postulated large break in the active recirculation loop, as discussed in NEDO-30060, "Vermont Yankee Nuclear Power Station Single Loop Operation." February 1983.

B. Linear Heat Generation Rate (LHGR)

Refer to the appropriate topical reports listed in Specification 6.7.A.4 for analyses methods.

C. Minimum Critical Power Ratio (MCPR)

Operating Limit MCPR

1. The MCPR operating limit is a cycle-dependent parameter which can be determined for a number of different combinations of operating modes, initial conditions, and cycle exposures in order to provide reasonable assurance against exceeding the Fuel Cladding Integrity Safety Limit (FCISL) for potential abnormal occurrences. The MCPR operating limits are justified by the analyses, the results of which are presented in the current cycle's Core Performance Analysis Report. Refer to the appropriate topical reports listed in Specification 6.7.A.4 for analysis methods. The 0.01 increase in MCPR operating limits for single loop operation accounts for increased core flow measurement and TIP reading uncertainties, as discussed in NEDO-30060, "Vermont Yankee Nuclear Power Station Single Loop Operation," February 1983.

BASES:

4.11 FUEL RODS

A. The APLHGR, LHGR and MCPR shall be checked daily to determine if fuel burnup, or control rod movement has caused changes in power distribution. Since changes due to burnup are slow, and only a few control rods are removed daily, a daily check of power distribution is adequate. For a limiting value to occur below 25% of rated thermal power, an unreasonably large peaking factor would be required, which is not the case for operating control rod sequences.

B. At certain times during plant startups and power changes the plant technical staff may determine that surveillance of APLHGR, LHGR and/or MCPR is necessary more frequently than daily. Because the necessity for such an augmented surveillance program is a function of a number of interrelated parameters, a reasonable program can only be determined on a case-by-case basis by the plant technical staff. The check of APLHGR, LHGR and MCPR will normally be done using the plant process computer. In the event that the computer is unavailable, the check will consist of either a manual calculation or a comparison of existing core conditions to those existing at the time of a previous check to determine if a significant change has occurred.

C. Minimum Critical Power Ratio (MCPR) - Surveillance Requirement

At core thermal power levels less than or equal to 25%, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience indicated that the resulting MCPR value is in excess of requirements by a considerable margin. With this low void content, any inadvertent core flow increase would only place operation in a more conservative mode relative to MCPR. During initial start-up testing of the plant, a MCPR evaluation will be made at 25% thermal power level with minimum recirculation pump speed. The MCPR margin will thus be demonstrated such that future MCPR evaluation below this power level will be shown to be unnecessary. The daily requirement for calculating MCPR above 25% rated thermal power is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR when a limiting control rod pattern is approached ensures that MCPR will be known following a change in power or power shape (regardless of magnitude) that could place operation at a thermal limit.

3.12 LIMITING CONDITIONS FOR OPERATION

3.12 REFUELING AND SPENT FUEL HANDLING

Applicability:

Applies to fuel handling, core reactivity limitations, and spent fuel handling.

Objective:

To assure core reactivity is within capability of the control rods, to prevent criticality during refueling, and to assure safe handling of spent fuel casks.

Specification:

A. Refueling Interlocks

The reactor mode switch shall be locked in the "Refuel" position during core alterations and the refueling interlocks, listed below, shall be operable except as specified in Specifications 3.12.D and 3.12.E.

1. Control Rod Blocks

   a. Mode switch in Startup/Hot Standby and refueling platform over the reactor.

   b. Fuel on any refueling hoist and refueling platform over the reactor.

   c. Mode switch in Refuel with one control rod withdrawal permit.

2. Refueling Platform Reverse Motion (toward reactor vessel) Block

   a. Mode switch in Startup/Hot Standby.

4.12 SURVEILLANCE REQUIREMENTS

4.12 REFUELING AND SPENT FUEL HANDLING

Applicability:

Applies to the periodic testing of those interlocks and instruments used during refueling and to the testing of the reactor building crane.

Objective:

To verify the operability of instrumentation and interlocks used in refueling and the operability of the reactor building crane.

Specification:

A. Refueling Interlocks

Prior to any fuel handling, with the Head off the reactor vessel, the refueling interlocks shall be functionally tested. They shall also be tested at weekly intervals thereafter until no longer required and following any repair work associated with the interlocks.

3.12 LIMITING CONDITIONS FOR OPERATION

b. Any control rod out and fuel on any refueling hoist.

3. Refueling Platform Hoists Block

a. Any control rod out and fuel on any refueling hoist over the vessel.

b. Hoist overload.

c. High position limitation.

B. Core Monitoring

During core alterations two SRM's shall be operable, one in the core quadrant where fuel or control rods are being moved and one in an adjacent quadrant. For an SRM to be considered operable the following conditions shall be satisfied.

1. The SRM shall be inserted to the normal operating level. (Use of special movable, dunking type detectors during initial fuel loading and major core alternations in place of normal detector is permissible as long as the detectors is connected into the proper circuitry which contain the required rod blocks).

2. The SRM shall have a minimum of 3 cps with all rods fully inserted in the core.

4.12 SURVEILLANCE REQUIREMENTS

B. Core Monitoring

Prior to making any alterations to the core the SRM's shall be functionally tested and checked for neutron response. Thereafter, the SRMs shall be checked daily for response.

3.12 LIMITING CONDITIONS FOR OPERATION

3. Prior to spiral unloading, the SRMs shall be proven operable as stated in Sections 3.12.B.1 and 3.12.B.2 above, however, during spiral unloading the count rate may drop below 3 cps.

4. Prior to spiral reloading, two diagonally adjacent fuel assemblies, which have previously accumulated exposure in the reactor, shall be loaded into their designated core positions next to each of the 4 SRMs to obtain the required 3 cps. Until these eight bundles have been loaded, the 3 cps requirement is not necessary.

C. <u>Fuel Storage Pool Water Level</u>

Whenever irradiated fuel is stored in the fuel storage pool the pool water level shall be maintained at a level of at least 36 feet.

4.12 SURVEILLANCE REQUIREMENTS

Prior to spiral unloading or reloading, the SRMs shall be functionally tested. Prior to spiral reloading, the SRMs shall be checked for neutron response.

C. <u>Fuel Storage Pool Water Level</u>

Whenever irradiated fuel is stored in the fuel storage pool, the pool level shall be recorded daily.

3.12 LIMITING CONDITIONS FOR OPERATION

D. Control Rod and Control Rod Drive Maintenance

A maximum of two non-adjacent control rods separated by more than two control cells in any direction, may be withdrawn from the core for the purpose of performing control rod and/or control rod drive maintenance provided the following conditions are satisfied:

1. The reactor mode switch shall be locked in the "Refuel" position. The refueling interlock which prevents more than one control rod from being withdrawn may be bypassed for one of the control rods on which maintenance is being performed. All other refueling interlocks shall be operable.

2. Specification 3.3.A.1 shall be met, or the control rod directional control valves for a minimum of eight control rods surrounding each drive out of service for maintenance shall be disarmed electrically and sufficient margin to criticality demonstrated.

3. SRMs shall be operable (a) in each core quadrant containing a control rod on which maintenance is being performed, and (b) in a quadrant adjacent to one of the quadrants specified in Specification 3.12.D.3.(a) above. Requirements for an SRM to be considered operable are given in Specification 3.12.B.

4.12 SURVEILLANCE REQUIREMENTS

D. Control Rod and Control Rod Drive Maintenance

1. Sufficient control rods shall be withdrawn prior to performing this maintenance to demonstrate with a margin of 0.25 percent Δk that the core can be made subcritical at any time during the maintenance with the strongest operable control rod fully withdrawn and all other operable rods fully inserted.

2. Alternately, if a minimum of eight control rods surrounding each control rod out of service for maintenance are to be fully inserted and have their directional control valves electrically disarmed, the 0.25 percent Δk margin shall be met with the strongest control rod remaining in service during the maintenance period fully withdrawn.

3.12 LIMITING CONDITIONS FOR OPERATION

E. Extended Core Maintenance

More than two control rods may be withdrawn from the reactor core provided the following conditions are satisfied:

1. The reactor mode switch shall be locked in the "Refuel" position. The refueling interlock which prevents more than one control rod from being withdrawn may be bypassed on a withdrawn control rod after the fuel assemblies in the cell containing (controlled by) that control rod have been removed from the reactor core. All other refueling interlocks shall be operable.

2. SRMs shall be operable in the core quadrant where fuel or control rods are being moved, and in an adjacent quadrant. The requirements for an SRM to be considered operable are given in Specification 3.12.B.

3. If the spiral unload/reload method of core alteration is to be used, the following conditions shall be met:

   a. Prior to spiral unload and reload, the SRMs shall be proven operable as stated in Specification 3.12.B1 and 3.12.B2. However, during spiral unloading, the count rate may drop below 3 cps.

4.12 SURVEILLANCE REQUIREMENTS

E. Extended Core Maintenance

Prior to control rod withdrawal for extended core maintenance, that control rods control cell shall be verified to contain no fuel assemblies.

1. This surveillance requirement is the same as that given in Specification 4.12.A.

2. This surveillance requirement is the same as that given in Specification 4.12.B.

3.12 LIMITING CONDITIONS FOR OPERATION

b. The core may be spirally reloaded to either the original configuration or a different configuration in the reverse sequence of that used to unload, with the exception that two (2) diagonally adjacent fuel assemblies, which have previously accumulated exposure in the reactor, shall be loaded into their designated core positions next to each of the four (4) SRMs to obtain the required 3 cps. Until these eight (8) bundles have been loaded, the 3 cps requirement is not necessary. Following insertion of the initial eight (8) bundles, the reactor will be spirally reloaded from the center cell outwards, until the core is fully loaded.

c. At least 50% of the fuel assemblies to be reloaded into the core shall have previously accumulated a minimum exposure of 1000 Mwd/T.

4.12 SURVEILLANCE REQUIREMENTS

3.12 LIMITING CONDITIONS FOR OPERATION

F. Fuel Movement

Fuel shall not be moved or handled in the reactor core for 24 hours following reactor shutdown to cold shutdown conditions.

G. Crane Operability

1. The Reactor Building crane shall be operable when the crane is used for handling of a spent fuel cask.

4.12 SURVEILLANCE REQUIREMENTS

F. Fuel Movement

Prior to any fuel handling or movement in the reactor core, the licensed operator shall verify that the reactor has been in the cold shutdown condition for a minimum of 24 hours.

G. Crane Operability

1. a. Within one month prior to spent fuel cask handling operations, an inspection of crane cables, sheaves, hook, yoke, and cask lifting trunnions will be made. These inspections shall meet the requirements of ANSI Standard B30.2, 1967. A crane rope shall be replaced if any of the replacement criteria given in ANSI B30.2.0-1967 are met.

   b. No-load mechanical and electrical tests will be conducted prior to lifting the empty cask from its transport vehicle to verify proper operations of crane controls, brakes and lifting speeds. A functional test of the crane brakes will be conducted each time an empty cask is lifted clear of its transport vehicle.

3.12 LIMITING CONDITIONS FOR OPERATION

2. Crane Travel

Spent fuel casks shall be prohibited from travel over irradiated fuel assemblies.

H. Spent Fuel Pool Water Temperature

Whenever irradiated fuel is stored in the spent fuel pool, the pool water temperature shall be maintained below 150°F.

4.12 SURVEILLANCE REQUIREMENTS

2. Crane Travel

Crane travel limiting mechanical stops shall be installed on the crane trolley rails prior to cask handling operations to prohibit cask travel over irradiated fuel assemblies.

H. Spent Fuel Pool Water Temperature

Whenever irradiated fuel is in the spent fuel pool, the pool water temperature shall be recorded daily. If the pool water temperature reaches 150°F, all refueling operations tending to raise the pool water temperature shall cease and measures taken immediately to reduce the pool water temperature below 150°F.

BASES:

3.12 & 4.12 REFUELING

A. During refueling operations, the reactivity potential of the core is being altered. It is necessary to require certain interlocks and restrict certain refueling procedures such that there is assurance that inadvertent criticality does not occur.

To minimize the possibility of loading fuel into a cell containing no control rod, it is required that all control rods are fully inserted when fuel is being loaded into the reactor core. This requirement assures that during refueling the refueling interlocks, as designed, will prevent inadvertent criticality. The core reactivity limitation of Specification 3.2 limits the core alterations to assure that the resulting core loading can be controlled with the Reactivity Control System and interlocks at any time during shutdown or the following operating cycle.

The addition of large amounts of reactivity to the core is prevented by operating procedures, which are in turn backed up by refueling interlocks on rod withdrawal and movement of the refueling platform. When the mode switch is in the "Refuel" position, interlocks prevent the refueling platform from being moved over the core if a control rod is withdrawn and fuel is on a hoist.

Likewise, if the refueling platform is over the core with fuel on a hoist, control rod motion is blocked by the interlocks. With the mode switch in the refuel position, only one control rod can be withdrawn.

B. The SRMs are provided to monitor the core during periods of station shutdown and to guide the operator during refueling operations and station startup. Requiring two operable SRMs in or adjacent to any core quadrant where fuel or control rods are being moved assures adequate monitoring of that quadrant during such alterations. The requirement of 3 counts per second provides assurance that neutron flux is being monitored. Under the special condition of complete spiral core unloading, it is expected that the count rate of the SRMs will drop below 3 cps before all the fuel is unloaded. Since there will be no reactivity additions, a lower number of counts will not present a hazard. When all of the fuel has been removed to the spent fuel storage pool, the SRMs will no longer be required. Requiring the SRMs to be operational prior to fuel removal assures that the SRMs are operable and can be relied on even when the count rate may go below 3 cps.

Prior to spiral reload, two diagonally adjacent fuel assemblies, which have previously accumulated exposure in the reactor, will be loaded into their designated core positions next to each of the 4 SRMs to obtain the required 3 cps. Exposed fuel continuously produces neutrons by spontaneous fission of certain plutonium isotopes, photo fission, and photo disintegration of deuterium in the moderator. This neutron production is normally great enough to meet the 3 cps minimum SRM requirement, thereby providing a means by which SRM response may be demonstrated before the spiral reload begins. During the spiral reload, the fuel will be loaded in the reverse sequence that it was unloaded with the exception of the initial eight (8) fuel assemblies which are loaded next to the SRMs to provide a means of SRM response.

BASES: 3.12 & 4.12 (Cont'd)

C. To assure that there is adequate water to shield and cool the irradiated fuel assemblies stored in the pool, a minimum pool water level is established. This minimum water level of 36 feet is established because it would be a significant change from the normal level, well above a level to assure adequate cooling (just above active fuel).

D. During certain periods, it is desirable to perform maintenance on two control rods and/or control rod drives at the same time. This specification provides assurance that inadvertent criticality does not occur during such maintenance.

The maintenance is performed with the mode switch in the "Refuel" position to provide the refueling interlocks normally available during refueling operations as explained in Part A of these Bases. In order to withdraw a second control rod after withdrawal of the first rod, it is necessary to bypass the refueling interlock on the first control rod which prevents more than one control rod from being withdrawn at the same time. The requirement that an adequate shutdown margin be demonstrated with the control rods remaining in service ensures that inadvertent criticality cannot occur during this maintenance. The shutdown margin is verified by demonstrating that the core is shut down even if the strongest control rod remaining in service is fully withdrawn. Disarming the directional control valves does not inhibit control rod scram capability.

E. The intent of this specification is to permit the unloading of a significant portion of the reactor core for such purposes as inservice inspection requirements, examination of the core support plate, etc. This specification provides assurance that inadvertent criticality does not occur during such operation.

This operation is performed with the mode switch in the "Refuel" position to provide the refueling interlocks normally available during refueling as explained in the Bases for Specification 3.12.A. In order to withdraw more than one control rod, it is necessary to bypass the refueling interlock on each withdrawn control rod which prevents more than one control rod from being withdrawn at a time. The requirement that the fuel assemblies in the cell controlled by the control rod be removed from the reactor core before the interlock can be bypassed ensures that withdrawal of another control rod does not result in inadvertent criticality. Each control rod essentially provides reactivity control for the fuel assemblies in the cell associated with that control rod. Thus, removal of an entire cell (fuel assemblies plus control rod) results in a lower reactivity potential of the core.

One method available for unloading or reloading the core is the spiral unload/reload. A spiral unloading pattern is one by which the fuel in the outermost cells (four fuel bundles surrounding a control rod) is removed first. Unloading continues by unloading the remaining outermost fuel by cell spiralling inward towards the center cell which is the last cell removed. Spiral reloading is reverse of unloading, with the exception that two (2) diagonally adjacent bundles, which have previously accumulated exposure in-core, are placed next to each of the 4 SRMs before the actual spiral reloading begins. The spiral reload then begins in the center cell and spirals outward until the core is fully loaded.

BASES: 3.12 & 4.12 (Cont'd)

Additionally, at least 50% of the fuel assemblies to be reloaded into the core shall have previously accumulated a minimum exposure of 1000 Mwd/T to ensure the presence of a minimum neutron flux as described in Bases Section 3.12.B.

F. The intent of this specification is to assure that the reactor core has been in the cold shutdown condition for at least 24 hours following power operation and prior to fuel handling or movement. The safety analysis for the postulated refueling accident assumed that the reactor had been shut down for 24 hours for fission product decay prior to any fuel handling which could result in dropping of a fuel assembly.

G. The operability requirements of the reactor building crane ensures that the redundant features of the crane have been adequately inspected just prior to using it for handling of a spent fuel cask. The redundant hoist system ensures that a load will not be dropped for any postulated credible single component failures. Details of the design of the redundant features of the crane and specific testing requirements for the crane are delineated in the Vermont Yankee document entitled "Reactor Building Crane Modification" (December 1975).

H. The Spent Fuel Pool Cooling System is designed to maintain the pool water temperature below 125°F during normal refueling operations. If the reactor core is completely discharged, the temperature of the pool water may increase to greater than 125°F. The RHR System supplemental fuel pool cooling may be used under these conditions to maintain the pool water temperature less than 150°F.

3.13 LIMITING CONDITIONS FOR OPERATION	4.13 SURVEILLANCE REQUIREMENTS

3.13 FIRE PROTECTION SYSTEM

Applicability:

Applies to the operational status of the fire protection systems.

Objective:

To assure adequate capability to detect and suppress a fire which could affect the safe shutdown of the reactor.

Specification:

A. Fire Detection

1. Except as specified in Specification 3.13.A.2 below, the minimum number of fire detection sensors and their associated instrument for each location shall be operable in accordance with Table 3.13.A.1, whenever the equipment it protects is required to be operable.

2. From and after the date that less than the minimum number of sensors or their associated instruments are found to be operable, a fire watch shall be established to inspect the location with the inoperable sensor or instruments at least once every hour. Restore the required number of sensors and instruments to operable status within 14 days or submit a report within the next 30 days to the Commission as specified in 6.7.C.2 outlining the cause of malfunction and the plans for restoring the instrument(s) to operable status.

4.14 FIRE PROTECTION SYSTEM

Applicability:

Applies to the surveillance requirements of the fire protection systems.

Objective:

To verify the operability of the fire protection systems.

Specification:

A. Fire Detection

1. Each of the sensors specified in 3.13.A.1 and their associated instruments including the supervisory circuitry shall be demonstrated operable at least once per 6 months.

3.13 LIMITING CONDITIONS FOR OPERATION

B. Vital Fire Suppression Water System

1. Except as specified in Specification 3.13.B.2 and 3.13.B.3 below, the Vital Fire Suppression Water System shall be operable with:

   a. Two fire pumps operable and lined up to the fire suppression loop.

   b. Water available from the Connecticut River.

   c. An operable flow path capable of taking suction from the Connecticut River and transferring the water through the distribution piping with operable sectionalizing control or isolation valves to the yard hydrant curb valves and the hose station isolation valves.

2. From and after the date that less than the above required equipment is operable, restore the component to operable status within 7 days or submit a report within the next 30 days to the Commission as specified in 6.7.C.2 outlining the plans and procedures to be used to provide for the loss of redundancy in this system.

3. With the fire suppression water supply system inoperable; and

   a. Establish a backup fire suppression water system within 24 hours,

4.13 SURVEILLANCE REQUIREMENTS

B. Vital Fire Suppression Water System

1. The Vital Fire Suppression Water System shall be demonstrated operable:

   a. At least once per month by starting each pump and operating it for 15 minutes.

   b. At least once each month by verifying each valve in the flow path is in its correct position. (For electrically supervised valves, adequate verification is a visual check of electrical indication.)

   c. At least once each year by performance of a system flush of the yard fire loop.

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

   e. At least once each 18 months:

      1) By performing a system functional test by simulating sequential automatic start of the fire pumps as applicable to maintain the Vital Fire Suppression Water System pressure of at least 125 psig.

3.13 LIMITING CONDITIONS FOR OPERATION

b. Submit a Special Report as specified in 6.7.C.2;

1) By telephone within 24 hours,

2) Confirmed by telegraph, mailgram, or facsimile transmission no later than the first working day following the event; and

3) In writing within 14 days following the event, outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to operable status, or

c. If a. above cannot be fulfilled, place the reactor in hot standby within the next six (6) hours and in cold shutdown with the following thirty (3) hours.

4.13 SURVEILLANCE REQUIREMENTS

2) By verifying that each pump will develop a flow of at least 2500 gpm at a discharge pressure of at least 115 psig corrected for river water level.

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

f. At least once per 3 years by performing a flow test in accordance with Chapter 5, Section II, of the Fire Protection Handbook, 14th Edition, published by the National Fire Protection Association.

2. The fire pump diesel engine shall be demonstrated OPERABLE:

a. At least once per month by verifying;

1) The fuel storage tank contains at least 150 gallons of fuel, and

2) The diesel starts from ambient conditions and operates for at least 20 minutes.

3.13 LIMITING CONDITIONS FOR OPERATION	4.13 SURVEILLANCE REQUIREMENTS
	b. At least once per quarter by verifying that a sample of diesel fuel from the fuel storage tank is within the acceptable limits specified in Table 1 of ASTM D975-68 with respect to viscosity, water content, and sediment. c. At least once per 18 months by verifying the diesel starts from ambient conditions on the auto-start signal and operates for $\geq$ 20 minutes while loaded with the fire pump. 3. The fire pump diesel starting 24-volt battery bank and charger shall be demonstrated OPERABLE: a. At least once per week by verifying that: 1) The electrolyte level of each battery is above the plates, and 2) The overall battery voltage is $\geq$ 24 volts. b. At least once per quarter by verifying that the specific gravity is appropriate for continued service of the battery. c. At least once per 18 months by verifying that:

3.13 LIMITING CONDITIONS FOR OPERATION

4.13 SURVEILLANCE REQUIREMENTS

1) The batteries, cell plates and battery racks show no visual indication of physical damage or abnormal deterioration, and

2) The battery-to-battery and terminal connections are clean, tight, free of corrosion and coated with anti-corrosion material.

C. Fire Hose Stations

1. Except as specified in 3.12.C.2 below, all hose stations inside the Reactor Building, Turbine Building, and those inside the Administration Building which provided coverage of the Control Room Building shall be operable whenever equipment in the areas protected by the fire hose stations is required to be operable.

2. With one or more of the fire hose stations specified in 3.13.C.1 above inoperable, route an additional equivalent capacity fire hose to the unprotected area(s) from an operable hose station within one hour.

C. Fire Hose Stations

1. Each fire hose station shall be verified to be operable:

   a. At least monthly by visual inspection of the station to assure all equipment is available.

   b. At least once each 18 months by removing the hose for inspection and replacing degraded coupling gaskets and reracking.

   c. At least once each year by hydro-statically testing each outside hose at 250 lbs.

   d. At least once per 3 years by hydro-statically testing inside hose at 150 lbs.

3.13 LIMITING CONDITIONS FOR OPERATION

D. High Pressure CO_2 System

1. Except as specified in Specification 3.13.D.2, the CO_2 systems located in the cable vault, switchgear room, and diesel fire pump day tank room shall be operable, whenever equipment in the area protected by the system is required to be operable.

2. From and after the date that the CO_2 system in the cable vault or a switchgear room is inoperable, within one hour a fire watch shall be established to inspect the location at least once every hour, provided that the fire detection system is operable in accordance with 3.13.A. If the fire detection system is also inoperable, within one hour a continuous fire watch shall be established with backup fire suppression equipment. Restore the CO_2 system to operable status within 14 days or submit a report within the next 30 days to the Commission as specified in 6.7.C.2 outlining the cause of inoperability and the plans for restoring the CO_2 system to operable status.

4.13 SURVEILLANCE REQUIREMENTS

e. At least once per 3 years, partially open hose station valves to verify valve operability and no blockage.

D. High Pressure CO_2 System

1. The CO_2 system located in the cable vault, switchgear room, and diesel fire pump day tank room shall be demonstrated operable.

   a. At least once per six months by verifying each CO_2 cylinder does not contain less than 90% of its initial charge.

   b. At least once per 18 months by verifying that the system, including associated ventilation dampers, will actuate automatically to a simulated actuation signal.

   c. At least once per operating cycle a flow path test shall be performed to verify flow through each nozzle.

3.13 LIMITING CONDITIONS FOR OPERATION

3. From and after the date that the CO_2 system in the diesel fire pump day tank room is inoperable, within one hour a fire watch shall be established to inspect the location at least once every hour. Restore the system to operable status within 14 days or submit a report within the next 30 days to the Commission as specified in 6.7.C.2 outlining the cause of inoperability and the plans for restoring the system to operable status.

E. Vital Fire Barrier Penetration Fire Seals

1. Except as specified in Specification 3.13.E.2 below, vital fire barrier penetration seals protecting the Reactor Building, Control Room Building, and Diesel Generator Rooms shall be intact.

2. From and after the date a vital fire barrier penetration fire seal is not intact, a continuous fire watch shall be established on at least one side of the affected penetration within 1 hour.

F. Sprinkler Systems

1. Except as specified in Specification 3.13.F.2 below, those sprinkler systems listed in Table 3.13.F.1 shall be operable whenever equipment in the area protected by those sprinklers is required to be operable.

4.13 SURVEILLANCE REQUIREMENTS

E. Vital Fire Barrier Penetration Fire Seals

1. Vital fire barrier penetration seals shall be verified to be functional by visual inspection at least once per operating cycle and following any repair.

F. Sprinkler Systems

1. Each of the sprinkler systems specified in Table 3.13.F.1 shall be demonstrated operable:

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

3.13 LIMITING CONDITIONS FOR OPERATION

2. From and after the date that one of the sprinkler systems specified in Table 3.13.F.1 is inoperable, a fire watch shall be established within one hour to inspect the location with the inoperable sprinkler system at least once every hour. Restore the sprinkler system to operable status within 14 days or submit a report within the next 30 days to the Commission as specified in 6.7.C.2 outlining the cause of the malfunction and the plans for restoring the system to operable status.

4.13 SURVEILLANCE REQUIREMENTS

b. At least once each month by verifying each valve in the flow path is in its correct position. (For electrically supervised valves, adequate verification is a visual check of electrical indication.)

c. At least once per 18 months by;

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

2. A visual inspection of the sprinkler headers to verify their integrity, and

3. A visual inspection of each nozzle's spray area to verify that the spray pattern is not obstructed.

4. Verifying that automatic valves actuate to their correct position from a test signal.

d. At least once per 3 years by performing a flow test through each open head sprinkler header and verifying each open head sprinkler nozzle is unobstructed.

3.13 LIMITING CONDITIONS FOR OPERATION

G. Foam Systems

1. Except as specified in Specification 3.13.G.2 below, the Recirculation M.G. Set Foam System shall be operable with its foam concentrate tank full (100 gallons) whenever the Recirculation M.G. Sets are operating.

2. From and after the date that the Recirculation M.G. Set Foam System is inoperable, a fire watch shall be established to inspect the location at least once every hour; and a foam nozzle shall be brought to the Reactor Building elevation containing the Recirculation M.G. Sets. A 100 gallon foam concentrate supply shall be available on site.

3. Except as specified in Specification 3.13.G.4 below, the Turbine Building Foam System shall be operable with its foam concentrate tank full (150 gallons).

4. From and after the date that the Turbine Building Foam System is inoperable a portable foam nozzle shall be brought to the Turbine Building Foam System location. A 150 gallon foam concentrate supply shall be available on-site.

4.13 SURVEILLANCE REQUIREMENTS

G. Foam Systems

1. The foam system specified in 3.13.G shall be demonstrated operable.

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

   b. At least once per 18 months by:

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

      2. A visual inspection of the foam system and equipment to verify integrity, and

      3. A visual inspection of the Recirculation M.G. Set Foam System foam nozzle area to verify that the spray pattern is not obstructed.

      4. Foam concentrate samples shall be taken and analyzed for acceptability.

3.13 LIMITING CONDITIONS FOR OPERATION	4.13 SURVEILLANCE REQUIREMENTS
	d. At least once per 3 years by performing an air flow test through the Recirculation M.G. Set foam header and verifying each foam nozzle is unobstructed.

TABLE 3.13.A.1

FIRE DETECTION SENSORS

	Sensor Location	Minimum No. of Sensors Required to Be Operable Heat	Flame	Smoke
1.	Cable Spreading Room & Station Battery Room	-	-	23
2.	Switchgear Room	-	-	20
3.	Diesel Generator Room (A)	-	-	2
4.	Diesel Generator Room (B)	-	-	2
5.	Intake Structure (Service Water)	1	1	1
6.	Recirc Motor Generator Set Area	3	-	8
7.a	Control Room Zone 1 (Control Room Ceiling)	-	-	14
7.b	Control Room Zone 2 (Control Room Panels)	-	-	18
7.c	Control Room Zone 3 (Control Room Panels)	-	-	25
7.d	Control Room Zone 4 (Control Room Panels)	-	-	10
7.e	Control Room Zone 5 (Exhaust & Supply Ducts)	-	-	2
8.a	Rx Bldg. Corner Rm NW 232	-	-	1
8.b	Rx Bldg. Corner Rm NW 213 (RCIC)	-	-	1
8.c	Rx Bldg. Corner Rm NE 232	-	-	1
8.d	Rx Bldg. Corner Rm NE 213	-	-	1
8.e	Rx Bldg. Corner Rm SE 232	-	-	1
8.f	Rx Bldg. Corner Rm SE 213	-	-	1
8.g	Rx Bldg. Corner Rm SW 232	-	-	1
9.	HPCI Room	-	-	8
10.	Torus area	12	-	16
11.	Rx Bldg. Cable Penetration Area	-	-	7
12.	Refuel Floor	-	-	13
13.	Diesel Oil Day Tank Room (A)	-	1*	1*
14.	Diesel Oil Day Tank Room (B)	-	1*	1*
15.	Turbine Loading Bay (vehicles)	-	3	-

*NOTE: The Diesel Day Tank Rooms require only one detector operable (1 flame or 1 smoke).

TABLE 3.13.F.1

SPRINKLER SYSTEMS

1. Reactor Building Penetration Area Preaction System
2. Diesel Generator Room A System
3. Diesel Generator Room B System
4. Turbine Loading Bay System
5. Diesel-driven Fire Pump System

BASES:

3.13 & 4.13 FIRE PROTECTION SYSTEMS

On May 11, 1976, Vermont Yankee received a letter from the NRC requesting that an in-depth evaluation of the existing fire protection systems be performed using Branch Technical Position (BTP) APCSB 9.5-1 as a guide. Concurrent with this evaluation a fire hazards analysis of the entire plant complex was required. In an effort to clarify the BTP an Appendix A was subsequently issued to specifically address operating plants. Enclosed with this Appendix the NRC requested that proposed Technical Specification on fire protection also be submitted. The subject section 3.13/4.13 and the following specific bases are those specifications evolving from these efforts.

A. The smoke, heat and flame detectors provide the early warning fire detection capability necessary to detect problems in vital areas of the plant. Surveillance requirements assure these sensors and their associated instruments to be operable. When the equipment protected by the detectors is not required to be operable, specifications covering the sensors and instruments do not apply.

B,C, D,F, and G The Vital Fire Suppression Water System, CO_2 systems, sprinkler systems and foam systems specifications are provided to meet pre-established levels of system operability in the event of a fire. These systems provide the necessary protection to assure safe reactor shutdown. Periodic surveillance testing provides assurance that vital fire suppression systems are operable.

E. Vital fire barrier penetration fire seals are provided to assure that the fire resistance rating of barriers is not reduced by a penetration. Surveillance inspections shall be performed to insure that the integrity of these seals is maintained.

The diesel fire pump has a design consumption rate of 18 gallons of fuel per hour; therefore, 150 gallons provides for greater than 8 hours of operation. Additional fuel can be delivered in about one hour and additional fuel is on site. When the equipment protected by the fire protection systems is not required to be operable, the specifications governing the fire protection system do not apply.

5.0 DESIGN FEATURES

5.1 Site

The station is located on the property on the west bank of the Connecticut River in the Town of Vernon, Vermont, which the Vermont Yankee Nuclear Power Corporation either owns or to which it has perpetual rights and easements. The site plan showing the exclusion area boundary, boundary for gaseous effluents and boundary for liquid effluents is on Figure 2.2-5 in the FSAR. The minimum distance to the boundary of the exclusion area as defined in 10CFR100.3 is 910 feet.

No part of the site shall be sold or leased and no structure shall be located on the site except structures owned by the Vermont Yankee Nuclear Power Corporation or related utility companies and used in conjunction with normal utility operations.

5.2 Reactor

A. The core shall consist of not more than 368 fuel assemblies.

B. The reactor core shall contain 89 cruciform-shaped control rods. The control material shall be boron carbide powder (B_4C) or hafnium, or a combination of the two.

5.3 Reactor Vessel

The reactor vessel shall be as described in Table 4.2-3 of the FSAR. The applicable design codes shall be as described in subsection 4.2 of the FSAR.

5.4 Containment

A. The principal design parameters and applicable design codes for the primary containment shall be as given in Table 5.2.1 of the FSAR.

B. The secondary containment shall be as described in subsection 5.3 of the FSAR and the applicable codes shall be as described in Section 12.0 of the FSAR.

C. Penetrations to the primary containment and piping passing through such penetrations shall be designed in accordance with standards set forth in subsection 5.2 of the FSAR.

5.5 Spent and New Fuel Storage

A. The new fuel storage facility shall be such that the effective multiplication factor (K_{eff}) of the fuel when dry is less than 0.90 and when flooded is less than 0.95.

B. The K_{eff} of the fuel in the spent fuel storage pool shall be less than or equal to 0.95.

C. Spent fuel storage racks may be moved (only) in accordance with written procedures which ensure that no rack modules are moved over fuel assemblies.

D. The number of spent fuel assemblies stored in the spent fuel pool shall not exceed 2870.

E. The maximum core geometry infinite lattice multiplication factor of any segment of the fuel assembly stored in the spent fuel storage pool or the new fuel storage facility shall be less than or equal to 1.31 at 20°C.

6.0 ADMINISTRATIVE CONTROLS

Administrative controls are the written rules, orders, instructions, procedures, policies, practices, and the designation of authorities and responsibilities by the management to obtain assurance of safety and quality of operation and maintenance of a nuclear power reactor. These controls shall be adhered to.

6.1 ORGANIZATION

A. Lines of authority, responsibility, and communication shall be established and defined for the highest management levels through intermediate levels to and including all operating organizational positions. These relationships shall be documented and updated, as appropriate, in the form of organizational charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements shall be documented in the Yankee Operational Quality Assurance Manual.

B. The Plant Manager shall be responsible for overall unit safe operation and shall have control over those on-site activities necessary for safe operation and maintenance of the plant. Succession to this responsibility during his absence shall be delegated in writing.

C. The Manager of Operations shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.

D. Conduct of operations of the plant will be in accordance with the following minimum conditions.

1. An individual qualified in radiation protection procedures shall be present on-site at all times when there is fuel in the reactor.

2. Minimum shift staffing on-site shall be in accordance with Table 6.1.1.

3. A dedicated, licensed Senior Operator shall be in charge of any reactor core alteration.

4. Qualifications with regard to educational background experience, and technical specialties of the key supervisory personnel listed below shall apply and be maintained in accordance with the levels described in the American National Standards Institute N18.1-1971, "Selection and Training of Personnel for Nuclear Power Plants".

   a. Plant Manager
   b. Superintendent(s)
   c. Chemistry Supervisor
   d. Radiation Protection Supervisor
   e. Operations Supervisor (See Item D.7, Page 190a)
   f. Reactor and Computer Supervisor
   g. Maintenance Supervisor
   h. Instrument and Control Supervisor
   i. Shift Supervisors

5. The Radiation Protection Supervisor or Plant Health Physicist shall meet or exceed the qualifications of Regulatory Guide 1.8, Revision 1 (September 1975).

6. The Shift Engineer shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design, and response and analysis of the plant for transients and accidents.

7. If the Operations Supervisor does not possess a Senior Operator License, then an Assistant Operations Supervisor shall be designated that does possess a Senior Operator License. All instructions to the shift crews involving licensed activities shall then be approved by designated Assistant Operations Supervisor.

8. The individuals who train the operating staff and those who carry out health physics and quality assurance functions may report to the appropriate on-site manager; however, they shall have sufficient organizational freedom to ensure their independence from operating pressures.

E. A Fire Brigade of at least 5 members shall be maintained on-site at all times.# This excludes 2 members of the minimum shift crew necessary for safe shutdown of the plant and any personnel required for other essential functions during a fire emergency.

Fire Brigade composition may be less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence of Fire Brigade members provided immediate action is taken to restore the Fire Brigade to within the minimum requirements.

TABLE 6.1.1

Vermont Yankee staff positions that shall be filled by personnel holding Senior Operator and Operator licenses are indicated in the following table:

Title	License Requirements
Operations Supervisor	Licensed Senior Operator (Except as specified in Specification 6.1.D.4)
Shift Supervisor	Licensed Senior Operator
Supervisory Control Room Operator	Licensed Senior Operator
Control Room Operator	Licensed Operator

	CONDITIONS	
MINIMUM SHIFT STAFFING ON-SITE	Plant Startup and Normal Operation (Note 1)	Cold Shutdown or Refueling With Fuel in the Reactor (Note 2)
Shift Supervisor	1	1
Supervisory Control Room Operator	1	-
Control Room Operator	2	1
Auxiliary Operator	2	1
Shift Engineer	1	-

NOTES:

(1) At least one Senior Licensed Operator and one Licensed Operator, or two Senior Licensed Operators, shall be in the Control Room.

(2) At least one Licensed Operator, or one Senior Licensed Operator, shall be in the Control Room.

6.2 REVIEW AND AUDIT

Organizational units for the review and audit of plant operations shall be constituted and have the responsibilities and authorities outlined below:

A. Plant Operations Review Committee

1. Membership

   a. Chairman: Plant Manager
   b. Vice-Chairman: Superintendent(s)
   c. Engineering Support Supervisor
   d. Operations Supervisor
   e. Maintenance Supervisor
   f. Reactor and Computer Supervisor
   g. Chemistry Supervisor
   h. Instrument and Control Supervisor
   i. Radiation Protection Supervisor

2. Qualifications

   The qualifications of the regular members of the Plant Operations Review Committee with regard to the combined experience and technical specialties of the individual members shall be maintained at a level at least equal to or higher than as described in Specification 6.1.

3. Meeting Frequency: Monthly, and as required, on call of the Chairman.

4. Quorum: Chairman or Vice-Chairman plus four members or their designated alternates.

   NOTE: For purposes of satisfying a quorum, a Vice-Chairman may be considered a member providing that Vice-Chairman is not presiding over the meeting.

5. Designated alternates shall be from other plant personnel in the appropriate disciplines or as selected by the Plant Manager; however, there shall be no more than three (3) alternates serving on the committee at any one time.

6. Responsibilities

   a. Review proposed normal, abnormal, and emergency operating procedures. Review all proposed maintenance procedures and proposed changes to those procedures; and any other proposed procedures or changes thereto as determined by the Plant Manager to affect nuclear safety.

   b. Review proposed tests and experiments.

   c. Review proposed changes to Technical Specifications.

   d. Review proposed changes or modifications to plant systems or equipment, which changes would require a change in procedures in (a) above.

   e. Review plant operations to detect any potential safety hazards.

f. Investigate reported instances of violations of Technical Specifications, such investigations to include reporting, evaluation, and recommendations to prevent recurrence, to the Manager of Operations.

g. Perform special reviews and investigations and render reports thereon as requested by the Chairman of the Nuclear Safety Audit and Review Committee.

7. Authority

a. The Plant Operation Review Committee shall be advisory.

b. The Plant Operation Review Committee shall recommend to the Plant Manager approval or disapproval of proposals under Items 6 (a) through (d) above.

1. In the event of disagreement between the recommendations of the Plant Operation Review Committee and the actions contemplated by the Plant Manager, the course determined by the Plant Manager to be the more conservative will be followed with immediate notification to the Manager of Operations.

c. The Plant Operation Review Committee shall make tentative determinations as to whether or not proposals considered by the Committee involve unreviewed safety questions. This determination shall be subject to review by the Nuclear Safety Audit and Review Committee.

8. Records

Minutes shall be kept at the plant of all meetings of the Plant Operation Review Committee and copies shall be sent to the Manager of Operations and the Nuclear Safety Audit and Review Committee.

B. Nuclear Safety Audit and Review Committee

1. The Committee shall consist of at least six (6) persons:

a. Chairman

b. Vice Chairman

c. Four technically qualified persons who are not members of the plant staff.

d. No more than three members shall be selected from the organization reporting to the Manager of Operations.

e. The Committee will obtain advice and counsel from scientific or technical personnel employed by the Company or other organizations whenever the Committee considers it necessary to obtain further scientific or technical assistance in carrying out its responsibilities.

f. The Committee membership and its Chairman and Vice Chairman shall be appointed as specified in the Yankee Quality Assurance Manual.

2. Qualifications

The Committee shall consist of a minimum of six (6) members of designated alternates who as a group employ expertise in the following areas:

a. Nuclear Power Plant Technology

b. Reactor Operations

c. Utility Operations

d. Power Plant Design

e. Reactor Engineering

f. Radiation Safety

g. Safety Analysis

h. Instrumentation and Control

i. Metallurgy

3. Meeting Frequency: Semi-annually and as required on call of the Chairman.

4. Quorum: Chairman or Vice Chairman plus four members or designated alternates.

5. Responsibilities:

a. Review proposed changes to the operating license including Technical Specifications.

b. Review minutes of meetings of the Plant Operation Review Committee to determine if matters considered by that committee involve unreviewed or unresolved safety questions.

c. Review the safety evaluations for changes to equipment or systems completed under the provisions of Section 50.59 10 CFR to verify that such actions did not constitute an unreviewed safety question.

d. Periodic audits of implementing procedures, shall be performed under cognizance of the Committee. Included in these audits, but not limited to, are the following specific activities:

i. plant operations;

ii. facility fire protection program;

iii. the radiological environmental monitoring program and the results thereof at least once per 12 months;

iv. the Off-Site Dose Calculation Manual and implementing procedures at least once per 24 months;

v. the Process Control Program and implementing procedures for processing and packaging of radioactive waste at least once per 24 months;

vi. the performance of activities required by the Quality Assurance Program to meet the provisions of Regulatory Guide 1.21, Revision 1, June 1974, and Regulatory Guide 4.1, Revision 1, April 1975, at least once per 12 months.

e. Investigate all reported instances of violations of Technical Specifications, reporting findings and recommendations to prevent recurrence to the Manager of Operations.

f. Perform special reviews and investigations and render reports thereon as requested by the Manager of Operations.

g. Review proposed tests and experiments and results thereof when applicable.

h. Review abnormal performance of plant equipment and anomalies.

i. Review unusual occurrences and incidents which are reportable under the provisions of 10 CFR Part 20 and 10 CFR Part 50.

j. Review of occurrences if safety limits are exceeded.

6. Authority

a. Review proposed changes to the operating license including Technical Specifications and revised bases for submittal to the NRC.

b. Review proposed changes or modifications to plant systems or equipment, provided that such changes or modifications do not involve unreviewed safety questions.

c. Recommend to the Manager of Operations appropriate action to prevent recurrence of any violations of Technical Specifications.

d. Evaluate actions taken by the Plant Operation Review Committee.

7. Records

Minutes of all meetings of this committee shall be recorded. Copies of the minutes shall be forwarded to the Manager of Operations, the Vice President - Operations, the Plant Manager and any others that the Chairman may designate.

6.3 ACTION TO BE TAKEN IN THE EVENT OF A REPORTABLE OCCURRENCE IN PLANT OPERATION

Applies to administrative action to be followed in the event of a reportable occurrence in plant operation.

Any reportable occurrence shall be reported to the Manager of Operations and shall be reviewed by the Plant Operations Review Committee. This Committee shall prepare a separate, sequentially numbered report for each reportable occurrence. Each report shall describe the circumstances leading up to and resulting from the occurrence, the corrective action taken by the shift, an attempt to define the cause of the occurrence, and shall recommend appropriate action to prevent or reduce the probability of a repetition of the occurrence.

Copies of all such reports shall be submitted to the Chairman of the Nuclear Safety Audit and Review Committee for review and to the Manager of Operations for review and approval of any recommendations.

6.4 ACTION TO BE TAKEN IF A SAFETY LIMIT IS EXCEEDED

Applies to administrative action to be followed in the event a safety limit is exceeded.

If a safety limit is exceeded, the reactor shall be shutdown immediately. An immediate report shall be made to the Manager of Operations. A complete analysis of the circumstances leading up to and resulting from the situation together with recommendations by the Plant Operations Review Committee shall also be prepared. This report shall be submitted to the Manager of Operations and the Chairman of the Nuclear Safety Audit and Review Committee.

Reactor operation shall not be resumed until authorized by the U.S. Nuclear Regulatory Commission.

6.5 PLANT OPERATING PROCEDURES

A. Detailed written procedures, involving both nuclear and non-nuclear safety, including applicable check-off lists and instructions, covering areas listed below shall be prepared and approved.

All procedures shall be adhered to.

1. Normal startup, operation and shutdown of systems and components of the facility.
2. Refueling operations.
3. Actions to be taken to correct specific and foreseen potential malfunctions of systems or components, suspected Primary System leaks and abnormal reactivity changes.
4. Emergency conditions involving potential or actual release of radioactivity.
5. Preventive and corrective maintenance operations which could have an effect on the safety of the reactor.
6. Surveillance and testing requirements.
7. Fire protection program implementation including minimum fire brigade requirements and training. The training program shall meet or exceed the requirements of Section 27 of the NFPA Code 1976. Training sessions will be scheduled as plant operations permit but will be completed in specified subjects annually. Initial fire brigade training shall be completed by March 13, 1978.

8. Process Control Program in-plant implementation.

9. Off-Site Dose Calculation Manual in-plant implementation.

B. Radiation control standards and procedures shall be prepared, approved and maintained and made available to all station personnel. These procedures shall show permissible radiation exposure, and shall be consistent with the requirements of 10 CFR Part 20. This radiation protection program shall be organized to meet the requirements of 10 CFR Part 20.

1. Paragraph 20.203, "Caution signs, labels, signals and controls". In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2), each high radiation area in which the intensity of radiation is 1000 mrem/hr or less shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit.* Any individual or group of individuals permitted to enter such areas shall be provided with one or more of the following:

   a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

   b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel have been made knowledgeable of them.

   c. A Health Physics qualified individual (i.e., qualified in radiation protection procedures) with a radiation dose rate monitoring device who is responsible for providing positive control over the activities within the area and who will perform periodic radiation surveillance at the frequency specified in the RWP. The surveillance frequency will be established by the Plant Health Physicist.

   The above procedure shall also apply to each high radiation area in which the intensity of radiation is greater than 1000 mrem/hr. In addition, locked doors shall be provided to prevent unauthorized entry into such areas and the keys shall be maintained under the administrative control of the Shift Supervisor on duty and/or the Plant Health Physicist.

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* Health Physics personnel shall be exempt from the RWP issuance requirement during the performance of their assigned radiation protection duties, providing they are following plant radiation protection procedures for entry into high radiation areas.

C. Procedures prepared for A and B above shall be reviewed and approved by the applicable Department Supervisor and the Plant Manager, or his designee (superintendent level).

D. Temporary changes to procedures described in Specification 6.5.A above which do not change the intent of the original procedure, may be made with the concurrence of two individuals holding senior operator licenses. Such changes shall be documented and subsequently reviewed by the PORC and approved by the Plant Manager or his designee.

E. Temporary changes to procedures described in Specification 6.5.B may be made with the concurrence of an individual holding a senior operator license and the health physicist on duty.

F. Licensed radioactive sealed sources shall be leak tested for contamination. Tests for leakage and/or contamination shall be performed by the licensee or by other persons specifically authorized by the Commission or an agreement state as follows:

1. Each licensed sealed source, except startup sources previously subjected to core flux, containing radioactive materials, other than Hydrogen 3, with half-life greater than thirty days and in any form, other than gas, shall be tested for leakage and/or contamination at intervals not to exceed six months.

2. The periodic leak test required does not apply to sealed sources that are stored and are not being used. The sources exempted from this test shall be tested for leakage prior to any use or transfer to another user unless they have been leak tested within six months prior to the date of use or transfer. In the absence of a certificate from a transferrer indicating that a leak test has been made within six months prior to the transfer, sealed sources shall not be put into use until tested.

3. Each sealed startup source shall be tested within 31 days prior to being subjected to core flux and following repair or maintenance to the source.

The leakage test shall be capable of detecting the presence of 0.005 microcurie of radioactive material on the test sample. If the test reveals the presence of 0.005 microcurie or more of removable contamination, it shall immediately be withdrawn from use, decontaminated, and repaired, or be disposed of in accordance with Commission regulations.

Notwithstanding the periodic leak tests required by this Technical Specification, any licensed sealed source is exempt from such leak test when the source contains 100 microcuries or less of beta and/or gamma emitting material or 5 microcuries or less of alpha emitting material.

A special report shall be prepared and submitted to the Commission within 90 days if source leakage tests reveal the presence of $\geq$0.005 microcuries of removable contamination.

6.6 PLANT OPERATING RECORDS

A. Records and/or logs relative to the following items shall be kept in a manner convenient for review and shall be retained for at least five years:

1. Records of normal plant operation, including power levels and periods of operation at each power level.

2. Records of principal maintenance activities, including inspection and repair or principal items of equipment pertaining to nuclear safety.

3. Records of reportable occurrences.

4. Records of periodic checks, inspection and/or calibrations performed to verify that surveillance requirements are being met.

5. Records of any special reactor test or experiments.

6. Records of changes made in the Operating Procedures.

7. Records of radioactive shipments.

8. Test results, in units of microcuries, for leak tests performed on licensed sealed sources.

9. Results of annual physical inventory verifying accountability of licensed sources on record.

B. Records and/or logs relative to the following items shall be recorded in a manner convenient for review and shall be retained for the life of the plant:

1. Records of substitution or replacement of principal items of equipment pertaining to nuclear safety.

2. Records of changes and drawing changes made to the plant as it is described in the Safety Analysis Report.

3. Records of plant radiation and contamination surveys.

4. Records of new and spent fuel inventory, transfer of fuel, and assembly histories.

5. Records of radioactivity in liquid and gaseous wastes released to the environment.

6. Records of radiation exposure for all plant personnel, including all contractors and visitors to the plant in accordance with 10 CFR 20.

7. Records of transient or operational cycling for those plant components that have been designed to operate safely for a limited number of transients or operational cycles.

8. Records of inservice inspections of the reactor coolant system.

9. Minutes of meetings of the Plant Operation Review Committee and the Nuclear Safety Audit and Review Board.

10. Records for Environmental Qualification which are covered under the provisions of paragraph 6.9.

11. Records of analysis required by the Radiological Environmental Monitoring Program.

6.7 REPORTING REQUIREMENTS

In addition to the applicable reporting requirements of Title 10 Code of Federal Regulations, the following identified reports shall be submitted to the Director of the appropriate Regional Office of Inspection and Enforcement unless otherwise noted.

A. Routine Reports

1. Startup Report

A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal or hydraulic performance of the plant. The report shall address each of the tests identified in the FSAR and shall, in general, include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption of commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the startup report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial power operation), supplementary reports shall be submitted at least every three months until all three events have been completed.

2. Annual Report

An annual report covering the previous calendar year shall be submitted prior to March 1 of each year. The annual report shall include a tabulation on an annual basis of the number of station, utility and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man-rem exposure according to work and job functions, 1/ e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling.

The dose assignment to various duty functions may be estimates based on pocket dosimeter, TLD or film badge measurement. Small exposures totaling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific major work functions.

3. Monthly Statistical Report

Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis to the Office of Management Information and Program Control, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the appropriate Regional Office, to arrive no later than the fifteenth of each month following the calendar month covered by the report. These reports shall include a narrative summary of operating experience during the report period which describes the operation of the facility.

4. Core Operating Limits Report

The core operating limits shall be established and documented in the Core Operating Limits Report (COLR) before each reload cycle or any remaining part of a reload cycle for the following: (a) The Average Planar Linear Heat Generation Rates (APLHGR) for Specifications 3.11.A and 3.6.G.1a, (b) The K_f core flow adjustment factor for Specification 3.11.C., (c) The Minimum Critical Power Ratio (MCPR) for Specifications 3.11.C and 3.6.G.1a, and (d) The Linear Heat Generation Rates (LHGR) for Specifications 2.1.A.1a, 2.1.B.1, and 3.11.B. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC in:

Report, E. E. Pilat, "Methods for the Analysis of Boiling Water Reactors Lattice Physics," YAEC-1232, December 1980 (Approved by NRC SER, dated September 15, 1982).

Report, D. M. VerPlanck, "Methods for the Analysis of Boiling Water Reactors Steady State Core Physics," YAEC-1238, March 1981 (Approved by NRC, SER, dated September 15, 1982).

Report, J. M. Holzer, "Methods for the Analysis of Boiling Water Reactors Transient Core Physics," YAEC-1239P, August 1981 (Approved by NRC SER, dated September 15, 1982).

Report, S. P. Schultz and K. E. St.John, "Methods for the Analysis of Guide Fuel Rod Steady-State Thermal Effects (FROSSTEY): Code/Model Description Manual," YAEC-1249P, April 1981 (Approved by NRC SER, dated September 27, 1985).

1/ This tabulation supplements the requirements of 20.407 of 10CFR Part 20.

Report, A. A. F. Ansari, "Methods for the Analysis of Boiling Water Reactors: Steady-State Core Flow Distribution Code (FIBWR)," YAEC-1234, December 1980 (Approved by NRC SER, dated September 15, 1982).

Report, S. P. Schultz and K. E. St.John, "Methods for the Analysis of Oxide Fuel Rod Steady-State Thermal Effects (FROSSTEY): Code Qualification and Application," YAEC-1265P, June 1981 (Approved by NRC SER, dated September 27, 1985).

Report, A. A. F. Ansari and J. T. Cronin, "Methods for the Analysis of Boiling Water Reactors: A System Transient Analysis Model (RETRAN)," YAEC-1233, April 1981. (Approved by NRC SERs, dated November 27, 1981 and September 4, 1984).

Report, A. A. F. Ansari, K. J. Burns and D. K. Beller, "Methods for the Analysis of Boiling Water Reactors: Transient Critical Power Ratio Analysis (RETRAN-TCPYA01)," YAEC-1299P, March 1982 (Approved by NRC SER, dated September 15, 1982).

Report, A. S. DiGiovine, et al., "CASMO-3G Validation," YAEC-1363-A, April 1988.

Report, A. S. DiGiovine, J. P. Gorski, and M. A. Tremblay, "SIMULATE-3 Validation and Verification," YAEC-1659-A, September 1988.

Report, R. A. Woehlke, et al., "MICBURN-3/CASMO-3/TABLES-3/SIMULATE-3 Benchmarking of Vermont Yankee Cycles 9 through 13," YAEC-1683-A, March 1989.

Report, J. T. Cronin, "Method for Generation of One-Dimensional Kinetics Data for RETRAN-02," YAEC-1694-A, June 1989.

Report, V. Chandola, M. P. LeFrancois, and J. D. Robichaud, "Application of One-Dimensional Kinetics to Boiling Water Reactor Transient Analysis Methods," YAEC-1693-A, Revision 1, November 1989.

Report, L. H. Steves, et. al, "HUXY: A Generalized Multirod Heatup Code with 10CFR50, Appendix K Heatup Option: User's Manual," XN-CC-33(A), Revision 1, dated November 14, 1975 (Approved by NRC SER, dated March 6, 1975).

Report, "RELAP5YA, A Computer Program for Light-Water Reactor System Thermal-Hydraulic Analysis," YAEC-1300P, October 1982 (Approved by NRC SERs, dated August 25, 1987 and October 21, 1992).

Report, R. T. Fernandez and H. C. daSilva, Jr., "Vermont Yankee BWR Loss-of-Coolant Accident Licensing Analysis Method," YAEC-1547, June 1986 (Approved by NRC SER, dated October 21, 1992).

Letter from R. W. Capstick (VYNPC) to USNRC, "HUXY Computer Code Information for the Vermont Yankee BWR LOCA Licensing Analysis Method," FVY 87-63, dated June 4, 1987 (Approved by NRC SER, dated February 27, 1991).

Letter from R. W. Capstick (VYNPC) to USNRC, "Request for Supplemental Safety Evaluation Report Supporting the Use of RELAP5YA for Vermont Yankee Nuclear Power Station," FVY 88-006, dated January 26, 1988 (Approved by NRC SERs, dated February 27, 1991 and October 21, 1992).

Letter from L. A. Tremblay, Jr. (VYNPC) to USNRC, "Supplementary Information Regarding NRC LOCA Analysis Review Effort," BVY 89-91, dated October 6, 1989 (Approved by NRC SER, dated October 21, 1992).

Letter from L. A. Tremblay, Jr. (VYNPC) to USNRC, "Supplementary Information Regarding NRC LOCA Analyses Review Effort," BVY 90-028, dated March 9, 1990 (Approved by NRC SER, dated October 21, 1992).

Letter from L. A. Tremblay, Jr. (VYNPC) to USNRC, "Response to Second Request for Additional Information on the Use of RELAP5YA," BVY 90-067, dated June 8, 1990 (Approved by NRC SER, dated February 27, 1991).

Letter from L. A. Tremblay, Jr. (VYNPC) to USNRC, "Response to Request for Additional Information on the Use of RELAP5YA," BVY 90-087, dated August 28, 1990 (Approved by NRC SER, dated October 21, 1992).

Letter from L. A. Tremblay, Jr. (VYNPC) to USNRC, "Response to Second Request for Additional Information on the Use of RELAP5YA," BVY 91-05, dated January 9, 1991 (Approved by NRC SER, dated October 21, 1992).

Letter from L. A. Tremblay, Jr. (VYNPC) to USNRC, "Response to Third Request for Additional Information on the Use of RELAP5YA," BVY 91-41, dated April 19, 1991 (Approved by NRC SER, dated October 21, 1992).

Letter from L. A. Tremblay, Jr. (VYNPC) to USNRC, "Supplementary Information Regarding the Use of RELAP5YA," BVY 92-12, dated February 7, 1992 (Approved by NRC SER, dated October 21, 1992).

Letter from R. W. Capstick (VYNPC) to USNRC, "Vermont Yankee LOCA Analysis Method FROSSTEY Fuel Performance Code (FROSSTEY-2)," FVY 87-116, dated December 16, 1987 (Approved by NRC SER, dated September 24, 1992.

Letter from R. W. Capstick (VYNPC) to USNRC, "Response to NRC Request for Additional Information on the FROSSTEY-2 Fuel Performance Code," BVY 89-65, dated July 14, 1989 (Approved by NRC SER, dated September 24, 1992).

Letter from R. W. Capstick (VYNPC) to USNRC, "Supplemental Information on the FROSSTEY-2 Fuel Performance Code," BVY 89-74, dated August 4, 1989 (Approved by NRC SER, dated September 24, 1992).

Letter from L. A. Tremblay, Jr. (VYNPC) to USNRC, "Responses to Request for Additional Information on FROSSTEY-2 Fuel Performance Code," BVY 90-045, dated April 19, 1990 (Approved by NRC SER, dated September 24, 1992).

Letter from L. A. Tremblay, Jr. (VYNPC) to USNRC, "Supplemental Information to VYNPC April 19, 1990 Response Regarding FROSSTEY-2 Fuel Performance Code," BVY 90-054, dated May 10, 1990 (Approved by NRC SER, dated September 24, 1992).

Letter from L. A. Tremblay, Jr. (VYNPC) to USNRC, "Responses to Request for Additional Information on FROSSTEY-2 Fuel Performance Code," BVY 91-024, dated March 6, 1991 (Approved by NRC SER, dated September 24, 1992).

Letter from L. A. Tremblay, Jr. (VYNPC) to USNRC, "LOCA-Related Responses to Open Issues on FROSSTEY-2 Fuel Performance Code," BVY 92-39, dated March 27, 1992 (Approved by NRC SER, dated September 24, 1992).

Letter from L. A. Tremblay, Jr. (VYNRC) to USNRC, "FROSSTEY-2 Fuel Performance Code - Vermont Yankee Response to Remaining Concerns," BVY 92-54, dated May 15, 1992 (Approved by NRC SER, dated September 24, 1992).

Report, "Loss-of-Coolant Accident Analysis for Vermont Yankee Nuclear Power Station," NEDO-21697, August 1977, as amended (Approved by NRC SER, dated November 30, 1977).

Report, "General Electric Standard Application for Reactor Fuel (GESTARII)," NEDE-24011-P-A, GE Company Proprietary (the latest NRC-approved version will be listed in the COLR).

The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met. The COLR, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

B. Reportable Occurrences

This section deleted.

C. Unique Reporting Requirements

1. Semiannual Effluent Release Report

a. Within 60 days after January 1 and July 1 of each year, a report shall be submitted covering the radioactive content of effluents released to unrestricted areas during the previous six months of operation.

b. The radioactive effluent release reports shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit as outlined in Regulatory Guide 1.21, Revision 1, June 1974, "Measuring, Evaluating and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants", with data summarized on a quarterly basis following the format of Appendix B thereof. For solid wastes the format for Table 3 in Appendix B of Regulatory Guide 1.21 shall be supplemented with three additional categories: class of solid wastes (as defined by 10CFR Part 61), type of container (e.g., LSA, Type A, Type B, Large Quantity), and solidification agent or absorbent, if any.

In addition, the radioactive effluent release report to be submitted 60 days after January 1 of each year shall include an annual summary of hourly meteorological data collected over the previous year. This annual summary may be either in the form of an hour-by-hour listing on magnetic tape of wind speed, wind direction, atmospheric stability, and precipitation (if measured), or in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability.* This same report (or a supplement to it to be submitted within 180 days of January 1 each year) shall include an assessment of the radiation doses due to the radioactive liquid and gaseous effluents released from the unit during the previous calendar year. (The semiannual effluent release report submitted within 60 days of July 1 each year need not contain any dose estimates from the previous 6 months' effluent releases.) The effluent reported submitted after January 1 each year shall also include an assessment of the radiation doses from radioactive effluents to member(s) of the public due to any allowed recreational activities inside the site boundary during the previous calendar year. All assumptions used in making these assessments (e.g., specific activity, exposure time and location) shall be included in these reports. For any batch or discrete gas volume releases, the meteorological conditions concurrent with the time of release of radioactive materials in gaseous effluents (as determined by sampling frequency and measurement) shall be used for determining the gaseous pathway doses. For radioactive materials released in continuous effluent streams, quarterly average meteorological conditions concurrent with the quarterly release period shall be used for determining the gaseous pathway doses. The assessment of radiation doses shall be performed in accordance with the Off-Site Dose Calculation Manual (ODCM).

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* In lieu of submission with the first half year radioactive effluent release report, the licensee has the options of retaining this summary of required meteorological data in a file that shall be provided to the NRC upon request.

With the limits of Specification 3.8.M.1 being exceeded during the calendar year, the radioactive effluent release report to be submitted 60 days after January 1 of each year shall also include an assessment of radiation doses to the likely most exposed real member(s) of the public from reactor releases (including doses from primary effluent pathways and direct radiation) for the previous calendar year to show conformance with 40CFR190, Environmental Radiation Protection Standards for Nuclear Power Operation.

The radioactive effluent release reports shall include a list and description of unplanned releases from the site to site boundary of radioactive materials in gaseous and liquid effluents made during the reporting period.

With the quantity of radioactive material in any outside tank exceeding the limit of Specification 3.8.D.1, describe the events leading to this condition in the next Radioactive Effluent Release Report.

If inoperable radioactive liquid effluent monitoring instrumentation is not returned to operable status prior to the next release pursuant to Note 4 of Table 3.9.1, explain in the next Radioactive Effluent Release Report the reason(s) for delay in correcting the inoperability.

If inoperable gaseous effluent monitoring instrumentation is not returned to operable status within 30 days pursuant to Note 5 of Table 3.9.2, explain in the next Radioactive Effluent Release Report the reason(s) for delay in correcting the inoperability.

With milk samples no longer available from one or more of the sample locations required by Table 3.9.3, identify the cause(s) of the sample(s) no longer being available, identify the new location(s) for obtaining available replacement samples, and include revised ODCM figure(s) and table(s) reflecting the new location(s) in the next Radioactive Effluent Release Report.

With a land use census identifying one or more locations which yield at least a 20 percent greater dose or dose commitment than the values currently being calculated in Specification 4.8.G.1, identify the new location(s) in the next Radioactive Effluent Release Report.

Changes made during the reporting period to the Process Control Program (PCP) and to the Off-Site Dose Calculation Manual (ODCM), shall be identified in the next Radioactive Effluent Release Report.

2. Special Reports

Special reports shall be submitted to the Director of the Office of Inspection and Enforcement Regional Office within the time period specified for each report.

a. Liquid Effluents, Specifications 3.8.B and 3.8.C

With the calculated dose from the release of radioactive materials in liquid effluents exceeding any of the limits of Specification 3.8.B.1, prepare and submit to the Commission within 30 days a special report which identifies the cause(s) for exceeding the limit(s) and defines the corrective actions taken to assure that subsequent releases will be in compliance with the limits of Specification 3.8.B.1.

With liquid radwaste being discharged without processing through appropriate treatment systems and estimated doses in excess of Specification 3.8.C.1, prepare and submit to the Commission within 30 days a special report which includes the following information:

(1) explanation of why liquid radwaste was being discharged without treatment, identification of any inoperable equipment or subsystems, and the reasons for the inoperability;

(2) action(s) taken to restore the inoperable equipment to operable status; and

(3) Summary description of action(s) taken to prevent a recurrence.

b. Gaseous Effluents, Specifications 3.8.F, 3.8.G, 3.8.H, and 3.8.I

With the calculated air dose from radioactive noble gases in gaseous effluents exceeding any of the limits of Specification 3.8.F.1, prepare and submit to the Commission within 30 days a special report which identifies the cause(s) for exceeding the limit(s) and the corrective action(s) taken to assure that subsequent releases will be in compliance with the limits of Specification 3.8.F.1. With the calculated dose from the release of Iodine-131, Iodine-133, tritium, and/or radionuclides in particulate form exceeding any of the limits of Specification 3.8.G.1, prepare and submit to the Commission within 30 days a special report which identifies the cause(s) for exceeding the limit(s) and the corrective action(s) taken to assure that subsequent releases will be in compliance with the limits of Specification 3.8.G.1.

With gaseous radwaste being discharged without processing through appropriate treatment systems as defined in Specification 3.8.H.1 for more than seven (7) consecutive days, or in excess of the limits of Specification 3.8.I.1, prepare and submit to the Commission within 30 days a special report which includes the following information:

(1) explanation of why gaseous radwaste was being discharged without treatment (Specification 3.8.H.1), or with resultant doses in excess of Specification 3.8.I.1, identification of any inoperable equipment or subsystems, and the reasons for the inoperability;

(2) action(s) taken to restore the inoperable equipment to operable status; and

(3) summary description of action(s) taken to prevent a recurrence.

c. <u>Total Dose, Specification 3.8.M</u>

With the calculated dose from the release of radioactive materials in liquid or gaseous effluents exceeding the limits of Specification 3.8.M, prepare and submit to the Commission within 30 days a special report which defines the corrective action(s) to be taken to reduce subsequent releases to prevent recurrence of exceeding the limits of Specification 3.8.M and includes the schedule for achieving conformance with these limits. This special report, required by 10CFR Part 20.405c, shall include an analysis that estimates the radiation exposure (dose) to a member of the public from station sources, including all effluent pathways and direct radiation, for the calendar year that includes the release(s) covered by this report. It shall also describe levels of radiation and concentrations of radioactive material involved, and the cause of the exposure levels or concentrations. If the estimated doses exceed any of the limits of Specification 3.8.M, and if the release condition resulting in violation of 40CFR Part 190 has not already been corrected, the special report shall include a request for a variance in accordance with the provisions of 40CFR Part 190. Submittal of the report is considered a timely request, and a variance is granted until staff action on the request is complete.

d. <u>Radiological Environmental Monitoring, Specification 3.9.C</u>

With the level of radioactivity as the result of plant effluents in an environmental sampling media at one or more of the locations specified in Table 3.9.3 exceeding the reporting levels of Table 3.9.4, prepare and submit to the Commission within 30 days from the receipt of the Laboratory Analyses a special report which includes an evaluation of any release conditions, environmental factors or other factors which caused the limits of Table 3.9.4 to be exceeded. 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 Surveillance Report.

e. Land Use Census, Specification 3.9.D

With a land use census not being conducted as required by Specification 3.9.D, prepare and submit to the Commission within 30 days a special report which identifies the reasons why the survey was not conducted, and what steps are being taken to correct the situation.

f. Vital Fire Protection System, Specification 3.13

Where required by Section 3.13, special reports shall be submitted to the Commission following the discovery of certain inoperable sensors, instruments, components, or systems in the vital fire protection system.

Note: Routine surveillance testing or design modification of sensors, instruments, components, or systems which lead to operation of sensors, instruments, components, or systems in a degraded mode do not require special reporting except where tests themselves reveal a degraded mode.

3. Environmental Radiological Monitoring

Radiological Environmental Surveillance Reports covering the operation of the unit during previous calendar year shall be submitted prior to May 1 of each year.

The annual Radiological Environmental Surveillance Report shall include summaries, interpretations, and an analysis of trends of the results of the radiological environmental surveillance activities for the report period, including a comparison with operational controls (as appropriate), and previous environmental surveillance reports and an assessment of the observed impact of the plant operation on the environment.

The annual Radiological Environmental Surveillance Report shall include summarized and tabulated results of all radiological environmental samples taken during the report period pursuant to the table and figures in the ODCM. In the event that some results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

With the level of radioactivity in an environmental sampling media at one or more of the locations specified in Table 3.9.3 exceeding the reporting levels of Table 3.9.4, the condition shall be described in the next annual Radiological Environmental Surveillance Report only if the measured level of radioactivity was not the result of plant effluents. With the radiological environmental monitoring program not being conducted as specified in Table 3.9.3, a description of the reasons for not conducting the program as required and the plans for preventing a recurrence shall be included in the next annual Radiological Environmental Surveillance Report.

The annual Radiological Environmental Surveillance Report shall also include the results of the land use census required by Specification 3.9.D. A summary description of the radiological environmental monitoring program including a map of all sampling locations keyed to a table giving distances and directions from the reactor shall be in the reports. If new environmental sampling locations are identified in accordance with Specification 3.9.D, the new locations shall be identified in the next annual Radiological Environmental Surveillance Report.

The reports shall also include a discussion of all analyses in which the LLD required by Table 4.9.3 was not achievable.

The results of license participation in the intercomparison program required by Specification 3.9.E shall be included in the reports. With analyses not being performed as required by Specification 3.9.E the corrective actions taken to prevent a recurrence shall be report to the Commission in the next annual Radiological Environmental Surveillance Report.

6.8 FIRE PROTECTION INSPECTION

A. An independent fire protection and loss prevention inspection and audit shall be performed annually utilizing either qualified off-site licensee personnel or an outside fire protection firm.

B. An inspection and audit by an outside fire consultant shall be performed at intervals no greater than 3 years.

6.9 ENVIRONMENTAL QUALIFICATION

A. By no later than June 30, 1982 all safety-related electrical equipment in the facility shall be qualified in accordance with the provisions of Division of Operating Reactors, "Guidelines for Evaluating Environmental Qualification of Class 1E Electrical Equipment in Operating Reactors" (DOR Guidelines); or NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment", December 1979. Copies of these documents are attached to Order for Modification of License DPR-28, dated October 24, 1980.

B. By no later than December 1, 1980, complete and auditable records must be available and maintained at a central location which describes the environmental qualification method used for all safety-related electrical equipment in sufficient detail to document the degree of compliance with the DOR Guidelines or NUREG-0588. Thereafter, such records should be updated and maintained current as equipment is replaced, further tested, or otherwise qualified.

6.10 INTEGRITY OF SYSTEMS OUTSIDE CONTAINMENT

A program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as low as practical levels will be implemented. This program shall include the following:

A. Provisions establishing preventive maintenance and periodic visual inspection requirements.

B. System leakage inspections, to the extent permitted by system design and radiological conditions, for each system at a frequency not to exceed refueling cycle intervals. The systems subject to this testing are: (1) Residual Heat Removal, (2) Core Spray, (3) Reactor Water Cleanup, (4) HPCI, (5) RCIC, and (6) Sampling Systems.

6.11 IODINE MONITORING

A program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas* under accident conditions will be implemented. This program shall include the following:

A. Training of personnel.

B. Procedures for monitoring.

C. Provisions for maintenance of sampling and analysis equipment.

6.12 PROCESS CONTROL PROGRAM (PCP)

A process control program shall contain the sampling, analysis, tests, and determinations by which wet radioactive waste from liquid systems is assured to be converted to a form suitable for off-site disposal.

A. Licensee initiated changes to the PCP:

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

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

   b. A determination that the change did not reduce the overall conformance of the dewatered spent resins/filter media waste product to existing criteria for solid waste shipments and disposal.

   c. Documentation of the fact that the change has been reviewed by PORC and approved by the Manager of Operations (MOO).

2. Shall become effective upon review by PORC and approval by the Manager of Operations (MOO).

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* Areas requiring personnel access for establishing hot shutdown conditions.

6.13 OFF-SITE DOSE CALCULATION MANUAL (ODCM)

An Off-Site Dose Calculation Manual shall contain the current methodology and parameters used in the calculation of off-site 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.

A. Licensee initiated changes to the ODCM:

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

a. Sufficiently detailed information to support the rationale for the change without benefit of additional or supplemental information. Information submitted should consist of a package of those pages of the ODCM which were changed with each page numbered and provided with the revision number, together with appropriate analyses or evaluations justifying the change(s).

b. A determination that the change will not reduce the accuracy or reliability of dose calculations or setpoint determinations.

c. Documentation of the fact that the change has been reviewed by PORC and approved by the Manager of Operations (MOO).

2. Shall become effective upon review by PORC and approved by the Manager of Operations (MOO).

6.14 MAJOR CHANGES TO RADIOACTIVE LIQUID, GASEOUS, AND SOLID WASTE TREATMENT SYSTEMS*

Licensee-initiated major changes to the radioactive waste systems (liquid, gaseous, and solid):

A. Shall be reported to the Commission in the semiannual Radioactive Effluent Release Report for the period in which the evaluation was reviewed by the PORC. The discussion of each change shall contain:

1. A summary of the evaluation that led to the determination that the change could be made in accordance with 10CFR Part 50.59;

2. Sufficient detailed information to support the reason for the change without benefit of additional or supplemental information;

3. A detailed description of the equipment, components, and processes involved and the interfaces with other plant systems;

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* Licensee may choose to submit the information called for in this Specification as part of the annual FSAR update.

4. An evaluation of the change, which shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the license application and amendments thereto;

5. An evaluation of the change, which shows the expected maximum exposures to member(s) of the public at the site boundary and to the general population that differ from those previously estimated in the license application and amendments thereto;

6. A comparison of the predicted releases of radioactive materials, in liquid and gaseous effluents and in solid waste, to the actual releases for the period prior to when the changes are to be made;

7. An estimate of the exposure to plant operating personnel as a result of the change; and

8. Documentation of the fact that the change was reviewed and found acceptable by PORC.

B. Shall become effective upon review and acceptance by PORC and approval by the Plant Manager.