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:

1. A Minimum Critical Power Ratio (MCPR) of less than 1.07 (1.09 for Single Loop Operation) shall constitute violation of the Fuel Cladding Integrity Safety Limit (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 Allowable Value (Run Mode)

When the mode switch is in the RUN position, the APRM flux scram Allowable Value shall be:

Two loop operation:

$S \leq 0.4W + 61.10\%$ for $0\% < W \leq 31.1\%$

$S \leq 1.28W + 33.31\%$ for $31.1\% < W \leq 54.0\%$

$S \leq 0.66W + 67.28\%$ for $54.0\% < W \leq 75.0\%$

With a maximum of 117.0% power for W > 75.0%

Single loop operation:

$S \leq 0.4W + 58.09\%$ for $0\% < W \leq 39.1\%$

$S \leq 1.28W + 23.56\%$ for $39.1\% < W \leq 61.9\%$

$S \leq 0.66W + 62.10\%$ for $61.9\% < W \leq 83.0\%$

With a maximum of 117.0% power for W > 83.0%

where:

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

## 1.1 SAFETY LIMIT

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

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

C. <u>Power Transient</u>

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

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

In the event of operation at >25% Rated Thermal Power the APRM gain shall be equal to or greater than 1.0.

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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 Allowable Value (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 are performed to demonstrate that the APRM flux scram over the range of settings from a maximum of 120% to the minimum flow biased setting provide protection from the fuel safety limit for all abnormal operational transients including those that may result in a thermal hydraulic instability.

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. The relationship between recirculation drive flow and reactor core flow is non-linear at low core flows. Due to stability concerns, separate APRM flow biased scram trip setting equations are provided for low core flows.

The APRM flow biased flux scram Allowable Value is the limiting value that the trip setpoint may have when tested periodically, beyond which appropriate action shall be taken. For Vermont Yankee, the periodic testing is defined as the calibration. The actual scram trip is conservatively set in relation to the Allowable Value to ensure operability between periodic testing. 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. The single loop

BASES: 2.1 (Cont'd)

operation equations are based on a bounding (maximum) difference between two loop and single loop drive flow at the same core flow of 8%.

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.

## 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 principal 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 specified in Table 2.2.1.

TABLE 2.2.1

Primary System Relief and Safety Valve Settings

| Number and Type of Valve(s) | Lift Setting[1] |
|---|---|
| 1 safety relief valve | 1080 psig |
| 2 safety relief valves | 1090 psig |
| 1 safety relief valve | 1100 psig |
| 3 safety valves | 1240 psig |

Note:

(1) As-left setpoint tolerance ±1%. As-found setpoint tolerance ±3%.

## 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. Deleted.

## 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. Deleted.

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 ACTIONS When Minimum Conditions For Operation Are Not Satisfied (3) |
|---|---|---|---|---|---|---|---|
| 1. | Mode Switch in Shutdown (5A-S1) | | X | X | X | 1 | A |
| 2. | Manual Scram (5A-S3A/B) | | X | X | X | 1 | A |
| 3. | IRM (7-41(A-F)) | | | | | | |
| | High Flux | ≤120/125 | X | X | | 2 | A |
| | INOP | | X | X | | 2 | A |
| 4. | APRM (APRM A-F) | | | | | | |
| | High Flux (flow bias) | Two loop operation: (4)<br>S≤ 0.4W+ 61.10% for 0% < W ≤ 31.1%<br>S≤ 1.28W+ 33.31% for 31.1% < W ≤ 54.0%<br>S≤ 0.66W+ 67.28% for 54.0% < W ≤ 75.0%<br>With a maximum of 117.0% power for W > 75.0%<br><br>Single loop operation: (4)<br>S≤ 0.4W+ 58.09% for 0% < W ≤ 39.1%<br>S≤ 1.28W+ 23.56% for 39.1% < W ≤ 61.9%<br>S≤ 0.66W+ 62.10% for 61.9% < W ≤ 83.0%<br>With a maximum of 117.0% power for W > 83.0% | | | X | 2 | A or B |
| | High Flux (reduced) | ≤15% | X | X | | 2 | A |
| | INOP | | | X | X | 2(5) | A or B |
| 5. | High Reactor Pressure (PT-2-3-55(A-D) (M) | ≤1055 psig | X | X | X | 2 | A |

TABLE 3.1.1

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT REQUIREMENTS
(Continued)

| | 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 ACTIONS When Minimum Conditions For Operation Are Not Satisfied (3) |
|---|---|---|---|---|---|---|---|
| 6. | High Drywell Pressure (PT-5-12(A-D)(M)) | ≤2.5 psig | X | X | X | 2 | A |
| 7. | Reactor Low (6) Water Level (LT-2-3-57A/B(M)) (LT-2-3-58A/B(M)) | ≥127.0 inches | X | X | X | 2 | A |
| 8. | Scram Discharge Volume High Level (LT-3-231(A-H)(M)) | ≤21 gallons | X | X | X | 2 (per volume) | A |
| 9. | Deleted | | | | | | |
| 10. | Main steamline isolation valve closure (POS-2-80A-A1,B1 POS-2-86A-A1,B1 POS-2-80B-A1,B2 POS-2-86B-A1,B2 POS-2-80C-A2,B1 POS-2-86C-A2,B1 POS-2-80D-A2,B2 POS-2-86D-A2,B2) | ≤10% valve closure | | | X | 4 | A or C |
| 11. | Turbine control valve fast closure (PS-(37-40)) | (9)(10) | | | X | 2 | A or D |
| 12. | Turbine stop valve closure (SVOS-5-(1-4)) | ≤10% valve(10) closure | | | X | 2 | A or D |

TABLE 3.1.1 NOTES (Cont'd)

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. The specified APRM High Flux scram (flow bias) Trip Setting is an Allowable Value, which is the limiting value that the trip setpoint may have when tested periodically. The actual scram trip setting is conservatively set in relation to the Allowable Value. "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.

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. Deleted.

8. Deleted.

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. Turbine stop valve closure and turbine control valve fast closure scram signals may be bypassed at $\leq$30% of reactor Rated Thermal Power.

11. Not used.

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 face adjacent or diagonally adjacent.

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) (7) | B | Heat Balance | Once Every 7 Days |
| Flow Bias | B | Standard Pressure and Voltage Source | Refueling Outage Every 3 Months (9) |
| LPRM (LPRM ND-2-1-104(80)) | B (5) | Using TIP System | Every 2,000 MWD/T average core exposure (8) |
| 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 |
| First Stage Turbine Pressure Permissive (PS-5-14(A-D)) | 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. Deleted.

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.

7. The IRM and SRM channels shall be determined to overlap during each startup after entering the STARTUP/HOT STANDBY MODE and the IRM and APRM channels shall be determined to overlap during each controlled shutdown, if not performed within the previous 7 days.

8. The specified frequency is met if the calibration is performed within 1.25 times the interval specified, as measured from the previous performance.

9. APRM trip unit calibration only.

BASES: 4.1 (Cont'd)

The calibration of the APRM High Flux Flow Bias trip units provides a check of the actual trip setpoints. If the trip setting is found to be less conservative than accounted for in the appropriate setpoint calculation, but is not beyond the Allowable Value specified in Table 3.1.1, the channel performance is still within the requirements of the plant safety analysis. However, if the trip setting is found to be less conservative than the Allowable Value specified in Table 3.1.1, the channel should be declared inoperable. Under these conditions, the setpoint should be readjusted to be equal to or more conservative than accounted for in the appropriate setpoint calculation. The specified trip unit calibration frequency (i.e., every 3 months) is consistent with the assumptions of the VYNPS setpoint methodology and the reliability analysis of NEDC-30851-P-A, "Technical Specification Improvement Analyses for BWR Reactor Protection System," March 1988.

LPRM gain settings are determined from the local flux profiles measured by the Traversing Incore Probe (TIP) System. This establishes the relative local flux profile for appropriate representative input to the APRM System. The 2,000 megawatt-days per short ton (MWD/T) frequency is based on operating experience with LPRM sensitivity changes, and that the resulting nodal power uncertainty, combined with other identified uncertainties, remains less than the total uncertainty (i.e., 8.7%) allowed by the GETAB safety limit analysis.

TABLE 3.2.5 NOTES

1. Deleted.

2. Deleted.

3. Deleted.

4. Deleted.

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. $\Delta W$ 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 and $\approx$ 8% for single loop operation.

6. Not used.

7. The trip may be bypassed when the reactor power is $\leq$30% of Rated Thermal Power. 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 the required number, place the inoperable channel in the tripped condition within one hour.

9. With one or two RBM channels inoperable:

   a. Deleted.

   b. If one RBM channel is inoperable, restore the inoperable channel to operable status within 24 hours; and

   c. If the required action and associated completion time of Note 9.b above is not met, or if two RBM channels are inoperable, place one RBM channel in the tripped condition within the next hour.

10. When a channel is placed in an inoperable status solely for performance of required surveillances, entry into associated Limiting Conditions for Operation and required action notes may be delayed for up to 6 hours provided the associated Trip Function maintains Control Rod Block initiation capability.

11. Deleted.

12. Required to be operable when the reactor mode switch is in the shutdown position.

13. With one or more Reactor Mode Switch - Shutdown Position channels inoperable, immediately suspend control rod withdrawal and immediately initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.

BASES: 3.2 (Cont'd)

control and/or bypass valves to open, resulting in a rapid depressurization and cooldown of the reactor vessel. The 800 psig trip 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 and temperature instrumentation is provided to detect a break in the HPCI and/or RCIC piping. The HPCI and RCIC steam supply pressure instrumentation is provided to isolate the systems when pressure may be too low to continue operation. These isolations are for equipment protection. However, they also provide a diverse signal to indicate a possible system break. These instruments are included in Technical Specifications because of the potential for possible system initiation failure if not properly tested. 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 Rod Block Monitor (RBM) control rod block functions are no longer credited in the Rod Withdrawal Error (RWE) Analysis. The RBM setpoints are based on providing operational flexibility in the MELLLA region.

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.

During hot shutdown, cold shutdown, and refueling when the reactor mode switch is required to be in the shutdown position, the core is assumed to be subcritical with sufficient shutdown margin; therefore, no positive reactivity insertion events are analyzed. The Reactor Mode Switch-Shutdown Position control rod withdrawal block, required to be operable with the mode switch in the shutdown position, ensures that the reactor remains subcritical by blocking control rod withdrawal, thereby preserving the assumptions of the safety analysis. Two channels are required to be

## 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. Deleted.

## 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. Deleted.

3.3 LIMITING CONDITIONS FOR OPERATION

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

1.1 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 |

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.

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 "General Electric Standard Application for Reactor Fuel (GESTAR II)," NEDE-24011-P-A, (the latest NRC-approved version will be listed in the COLR).

5. The Source Range Monitor (SRM) system provides a scram function in noncoincident configuration. 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. Deleted.

## 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 when the reactor mode switch is in either the "Startup/Hot Standby" or "Run" position, except to allow testing of instrumentation associated with the reactor mode switch interlock functions provided:

1. Reactor coolant temperature is less than or equal to 212° F;
2. All control rods remain fully inserted in core cells containing one or more fuel assemblies; and
3. No core alterations are in progress.

## 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 1320 psig shall be verified for each pump.
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:

3. Testing that the setting of the pressure relief valves is between 1400 and 1490 psig.
4. 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 vessel is available. Both loops shall be tested over the course of two operating cycles.

BASES:

3.4 & 4.4 REACTOR STANDBY LIQUID CONTROL SYSTEM

A. Normal Operation

The design objective of the Reactor Standby Liquid Control System (SLCS) 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 Standby Liquid Control System is designed to inject a quantity of boron which produces a concentration of 800 ppm of natural boron in the reactor core in less than 138 minutes. An 800 ppm natural boron concentration in the reactor core is required to bring the reactor from full power to a 5% $\Delta k$ 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% natural 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.

In addition to its original design basis, the Standby Liquid Control System also satisfies the requirements of 10CFR50.62(c)(4) on anticipated transients without scram (ATWS) by using enriched boron. The ATWS rule adds hot shutdown and neutron absorber (i.e., boron-10) injection rate requirements that exceed the original Standby Liquid Control System design basis. However, changes to the Standby Liquid Control System as a result of the ATWS rule have not invalidated the original design basis.

With the reactor mode switch in the "Run" or "Startup/Hot Standby" position, shutdown capability is required. With the mode switch in "Shutdown," control rods are not able to be withdrawn since a control rod block is applied. This provides adequate controls to ensure that the reactor remains subcritical. With the mode switch in "Refuel," only a single control rod can be withdrawn from a core cell containing fuel assemblies. Determination of adequate shutdown margin by Specification 3.3.A ensures that the reactor will not become critical. Therefore, the Standby Liquid Control System is not required to be operable when only a single control rod can be withdrawn.

Pump operability testing (by recirculating demineralized water to the test tank)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. The maximum SLCS pump discharge pressure during the limiting ATWS event is 1320 psig. This value is based on a peak reactor vessel lower plenum pressure of 1290 psia that occurs during the limiting ATWS event at the time of SLCS initiation, i.e., 120 seconds into the event. There is adequate margin to prevent the SLCS relief valve from lifting. With a nominal SLCS relief valve setpoint of 1400 psig, there is a margin of 80 psi between the peak SLCS pump discharge pressure and the relief valve nominal setpoint. Recirculation of the borated solution is done during each operating cycle to ensure one suction line from the boron tank is clear. In addition, at least once during each operating cycle, one of the standby liquid control loops will be initiated to verify that a flow path from a pump to the reactor vessel is available by pumping demineralized water into the reactor vessel.

BASES: 3.4 & 4.4 (Cont'd)

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

C. Standby Liquid Control System Tank - Borated Solution

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 3.4.2. Temperature and liquid level alarms for the system are annunciated in the Control Room.

Once the solution has been made up, boron concentration will not vary unless more boron or 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 personnel.

Boron concentration, solution temperature, and volume are checked on a frequency to assure a high reliability of operation of the system should it ever be required. Isotopic tests of the sodium pentaborate are performed periodically to ensure that the proper boron-10 atom percentage is being used.

10CFR50.62(c)(4) requires a Standby Liquid Control System with a minimum flow capacity and boron content equivalent to 86 gpm of 13 weight percent natural sodium pentaborate solution in the 251-inch reactor pressure vessel reference plant. Natural sodium pentaborate solution is 19.8 atom percent boron-10. The relationship expressed in Specification 3.4.C.3 also contains the ratio M251/M to account for the difference in water volume between the reference plant and Vermont Yankee. (This ratio of masses is 628,300 lbs./401,247 lbs.)

To comply with the ATWS rule, the combination of three Standby Liquid Control System parameters must be considered: boron concentration, Standby Liquid Control System pump flow rate, and boron-10 enrichment. Fixing the pump flow rate in Specification 3.4.C.3 at the minimum flow rate of 35 gpm conservatively establishes a system parameter that can be used in satisfying the ATWS requirement, as well as the original system design basis. If the product of the expression in Specification 3.4.C.3 is equal to or greater than unity, the Standby Liquid Control System satisfies the requirements of 10CFR50.62(c)(4).

## 3.6 LIMITING CONDITIONS FOR OPERATION

### D. Safety and Relief Valves

1. During reactor power operating conditions and whenever the reactor coolant pressure is greater than 150 psig and temperature greater than 350°F, all safety valves and at least three of the four relief valves shall be operable.

2. If Specification 3.6.D.1 is not met, initiate an orderly shutdown and the reactor coolant pressure shall be below 150 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.

   Inservice inspection of piping, identified in NRC Generic Letter 88-01, shall be performed in accordance with the staff positions on schedule, methods, and personnel and sample expansion included in the Generic Letter or in accordance with alternate measures approved by NRC Staff.

3.11 LIMITING CONDITIONS FOR OPERATION

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 operation at ≥25% Rated Thermal Power, the APLHGR for each type of fuel as a function of average planar exposure, power, and flow 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 operation at ≥25% Rated Thermal Power it is determined by normal surveillance that the limiting value for APLHGR is being exceeded, APLHGR(s) shall be returned to within prescribed limits within two (2) hours; otherwise, the reactor shall be brought to <25% Rated Thermal Power within 4 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

4.11 SURVEILLANCE REQUIREMENTS

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, power, and flow shall be determined once within 12 hours after ≥25% Rated Thermal Power and daily during operation at ≥25% Rated Thermal Power thereafter.

## 3.11 LIMITING CONDITIONS FOR OPERATION

### C. Minimum Critical Power Ratio (MCPR)

1. During operation at ≥25% Rated Thermal Power 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. If at any time during operation at ≥25% Rated Thermal Power it is determined by normal surveillance that the limiting value for MCPR is being exceeded, MCPR(s) shall be returned to within the prescribed limits within two (2) hours; otherwise, the reactor power shall be brought to <25% Rated Thermal Power within 4 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 once within 12 hours after ≥25% Rated Thermal Power and daily during operation at ≥25% Rated Thermal Power thereafter.

BASES:

3.11 FUEL RODS

A. Average Planar Linear Heat Generation Rate (APLHGR)

Refer to the appropriate topical reports listed in Specification 6.6.C 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 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. See Core Operating Limits Report for the cycle-specific reduction factor.

Flow dependent MAPLHGR limits, MAPFAC(F), were designed to assure adherence to all fuel thermal-mechanical design bases. The same transient events used to support the MCPR(F) operating limits were analyzed, and the resulting overpowers were statistically evaluated as a function of the initial and maximum core flow. From the bounding overpowers, the MAPFAC(F) limits were derived such that the peak transient LHGR would not exceed fuel mechanical limits. The flow-dependent MAPLHGR limits are cycle-independent and are specified in terms of multipliers, MAPFAC(F), to be applied to the rated MAPLHGR values.

Power-dependent MAPLHGR limits, expressed in terms of a MAPLHGR multiplier, MAPFAC(P), are substituted to assure adherence to the fuel thermal-mechanical design bases. Both incipient centerline melting of fuel and plastic strain of the cladding are considered in determining the power dependent MAPLHGR limit. Generally, the limiting criterion is incipient centerline melting. The power-dependent MAPFAC(P) multipliers were generated using the same database as used to determine the MCPR multiplier (Kp). Appropriate MAPFAC(P) multipliers are selected based on plant-specific transient analyses with suitable margin to assure applicability to future reloads. These limits are derived to assure that the peak transient MAPLHGR for any transient is not increased above the fuel design bases values.

B. Linear Heat Generation Rate (LHGR)

Refer to the appropriate topical reports listed in Specification 6.6.C for analyses methods.

Power and flow dependent LHGR limits are implemented using LHGRFAC multipliers on the standard LHGR limits. The LHGRFAC multipliers are identical to the MAPFAC multipliers.

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 Supplemental Reload

BASES:

3.11 FUEL RODS (Continued)

Licensing Report. Refer to the appropriate topical reports listed in Specification 6.6.C for analysis methods. The increase in MCPR operating limits for single loop operation accounts for increased core flow measurement and TIP reading uncertainties.

Flow-dependent MCPR limits, MCPR(F), are necessary to assure that the Safety Limit MCPR (SLMCPR) is not violated during recirculation flow increase events. The design basis flow increase event is a slow (maximum two pump runout rate of 1%/second) recirculation flow increase event which is not terminated by scram, but which stabilizes at a new core power corresponding to the maximum possible core flow. Flow runout events were analyzed along a constant xenon, constant feedwater temperature flow control line assuming a quasi steady-state plant heat balance. The ARTS-based MCPR(F) limit is specified as an absolute value and is cycle-independent. The operating limit is based on the maximum core flow limiter setting of 109.5% in the Recirculation Flow Control System.

Above the power at which the scram is bypassed ($P_{bypass}$), bounding power-dependent trend functions have been developed. This trend function, Kp, is used as multiplier to the rated MCPR operating limits to obtain the power-dependent MCPR limits, MCPR(P). Below the power at which the scram is automatically bypassed (Below $P_{bypass}$), the MCPR(P) limits are actual absolute Operating Limit MCPR (OLMCPR) values, rather than multipliers on the rated power OLMCPR.

BASES:

4.11 FUEL RODS

A. The APLHGR, LHGR and MCPR shall be checked daily when operating at $\geq$25% Rated Thermal Power 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. The 12 hour allowance after thermal power $\geq$25% Rated Thermal Power is achieved is acceptable given the large inherent margin to operating limits at low power levels.

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.

If a reactor power distribution limit is exceeded, an assumption regarding an initial condition of the DBA analysis, transient analyses, or the fuel design analysis may not be met. Therefore, prompt action should be taken to restore the APLHGR, LHGR or MCPR to within the required limits such that the plant operates within analyzed conditions and within design limits of the fuel rods. The 2 hour completion time is sufficient to restore the APLHGR, LHGR, or MCPR to within its limits and is acceptable based on the low probability of a transient or DBA occurring simultaneously with the APLHGR, LHGR, or MCPR out of specification.

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 during normal operation.

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 Self-Reading Dosimeter (SRD), 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 should be assigned to specific major work functions.

B. Monthly Operating Reports

Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis 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.

C. 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:

1. The Average Planar Linear Heat Generation Rates (APLHGR) for Specifications 3.11.A and 3.6.G.1a,
2. The Minimum Critical Power Ratio (MCPR) for Specifications 3.11.C and 3.6.G.1a,
3. The Linear Heat Generation Rates (LHGR) for Specifications 2.1.A.1a and 3.11.B, and
4. The Power/Flow Exclusion Region for Specifications 3.6.J.1a and 3.6.J.1b.

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).

[1] This tabulation supplements the requirements of 20.2206 of 10 CFR Part 20.