PROPOSED TECH SPEC

TS 3.1/4.1

"REACTOR PROTECTION SYSTEM"

3.1/4.1 REACTOR PROTECTION SYSTEM
SPECIFICATIONS

LIMITING CONDITIONS FOR OPERATION

A. Reactor Protection System

The reactor protection system instrumentation CHANNELS shall be OPERABLE with the setpoints, minimum number of TRIP SYSTEMS and minimum number of instrument CHANNELS as shown in Table 3.1-1. The system response times from the opening of the sensor contact up to and including the opening of the trip actuator contacts shall not exceed 50 milliseconds.

APPLICABILITY:

As shown in Table 3.1-1.

ACTION:

1. With a reactor protection system instrumentation setpoint less conservative than the value shown in the Trip Level Setting column of Table 3.1-1, declare the CHANNEL inoperable and follow ACTION 3.1.A.2 or 3.1.A.3 below until the CHANNEL is restored to OPERABLE status with its setpoint adjusted consistent with the Trip Level Setting value.

2. With the number of OPERABLE CHANNELS less than required by the Minimum OPERABLE CHANNELS per TRIP SYSTEM requirement for one TRIP SYSTEM, place the inoperable CHANNEL(s) and/or that TRIP SYSTEM in the tripped

SURVEILLANCE REQUIREMENTS

A. Reactor Protection System

Surveillance of the reactor protection system instrumentation CHANNELS shall be performed as follows:

1. Reactor protection instrumentation systems shall be functionally tested, calibrated and checked as indicated in Tables 4.1-1 and 4.1-2.

2. The system response times for each Trip Function shown in Table 3.1-1 shall be demonstrated to be within its limit at least each REFUELING OUTAGE. Each test shall include at least one CHANNEL per TRIP SYSTEM such that all CHANNELS are tested at least once every (N) REFUELING OUTAGES where (N) is the total number of redundant CHANNELS in a specific reactor TRIP SYSTEM.

condition within 12 hours. An inoperable CHANNEL need not be placed in the tripped condition when this would cause the PROTECTIVE FUNCTION to occur. In these cases, the inoperable CHANNEL shall be restored to OPERABLE status within 6 hours after the CHANNEL was first determined to be inoperable or the ACTION required by Table 3.1-1 shall be entered.

3. With the number of OPERABLE CHANNELS less than required by the Minimum OPERABLE CHANNELS per TRIP SYSTEM requirement for both TRIP SYSTEMS, place at least one TRIP SYSTEM in the tripped condition within 1 hour and then take the ACTION required by Table 3.1-1. The TRIP SYSTEM need not be placed in the tripped condition if this would cause the PROTECTIVE FUNCTION to occur. When a TRIP SYSTEM can be placed in the tripped condition without causing the PROTECTIVE FUNCTION to occur, place the TRIP SYSTEM with the most inoperable CHANNELS in the tripped condition; if both systems have the same number of inoperable CHANNELS, place either TRIP SYSTEM in the tripped condition.

B. APRM Scram and Control Rod Block Flow Biased Upscale Setpoints

The APRM flow biased neutron flux upscale scram trip setpoint and flow biased neutron flux upscale

B. APRM Scram and Control Rod Block Flow Biased Upscale Setpoints

The core power distribution shall be checked daily for MFLPD and compared with the FRP.

control rod block trip setpoint shall be established according to the equations in Specifications 2.1.A.1 and 2.1.B.

APPLICABILITY:

OPERATIONAL MODE 1, when thermal power is greater than or equal to 25% of RATED THERMAL POWER.

ACTION:

1. With the APRM flow biased neutron flux upscale scram trip setpoint and/or the flow biased neutron flux upscale control rod block trip setpoint less conservative than the value shown in the equations in Specifications 2.1.A.1 and 2.1.B, initiate corrective action within 15 minutes and within 6 hours, adjust the setpoints to be consistent with the Trip Setpoint values or increase the APRM gain as described in Specification 2.1.A.1 and 2.1.B or reduce thermal power to less than 25% of RATED THERMAL POWER within the next 4 hours.

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TABLE 3.1-1

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENTS

Trip Function	Minimum OPERABLE CHANNELS Per TRIP SYSTEM (a)(b)	Trip Level Setting	Applicable OPERATIONAL MODES	ACTION
1. Mode Switch in Shutdown	1 1 1	N.A.	1, 2 3, 4 5	1 7 2
2. Manual Scram	1 1 1	N.A.	1, 2 3, 4 5	1 7 6
3. IRM (c)				
a. High Flux	3 2 3(n)	≤ 120/125 of full scale	2 3, 4 5(m)	1 7 2
b. Inoperative	3 2 3(n)	N.A.	2 3, 4 5	1 7 2
4. APRM (f)				
a. High Flux (flow biased)	2	Tech Spec 2.1.A.1	1	3
b. Inoperative	2 2 2(n)	N.A.	1, 2 3 5(m)	1 7 2
c. High Flux (15% scram)	2 2 2(n)	Tech Spec 2.1.A.2	2 3 5(m)	1 7 2
d. High Flux (Scram Clamp)	2	Tech Spec 2.1.A.1	1	3
5. Reactor High Pressure	2	≤ 1060 psig	1, 2(g)	1
6. Drywell High Pressure	2	≤ 2.5 psig	1, 2(h)	1
7. Reactor Low Water Level	2	≥ 8 inches (d)	1, 2	1

TABLE 3.1-1

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENTS

Trip Function	Minimum OPERABLE CHANNELS Per TRIP SYSTEM (a)(b)	Trip Level Setting	Applicable OPERATIONAL MODES	ACTION
8. Scram Discharge Volume High Water Level	2/bank 2/bank	≤ 40 gallons	1, 2 5(i)(l)	1 2
9. Turbine Condenser Low Vacuum	2	≥ 21 inches Hg vacuum	1	4
10. Main Steam Line High Radiation (e)	2	≤ 15 X Normal Full Power Background	1, 2(g)	4
11. Main Steam Line Isolation Valve Closure	4 (k)	≤ 10% Valve Closure	1	3
12. Turbine Control Valve Fast Closure	2	≥ 460 psig (o)	1(j)	5
13. Turbine Stop Valve Closure	4	≤ 10% Valve Closure	1(j)	5
14. Turbine EHC Control Fluid Low Pressure	2	≥ 900 psig	1(j)	5

TABLE 3.1-1 (Continued)

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENTS

ACTIONS

ACTION 1 - Be in at least HOT SHUTDOWN within 12 hours.

ACTION 2 - Suspend all operations involving CORE ALTERATIONS* and insert all insertable control rods within one hour.

ACTION 3 - Be in at least STARTUP within 8 hours.

ACTION 4 - Be in STARTUP with the main steam line isolation valves closed within 8 hours or in at least HOT SHUTDOWN within 12 hours.

ACTION 5 - Initiate a reduction in thermal power within 15 minutes and reduce turbine first stage pressure to that which corresponds to less than 45% of rated steam flow, within 2 hours.

ACTION 6 - Suspend all operations involving CORE ALTERATIONS*, and insert all insertable control rods and lock the reactor mode switch in the SHUTDOWN position within one hour.

ACTION 7 - Verify all insertable control rods to be inserted in the core and lock the reactor mode switch in the Shutdown position within one hour.

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* Except replacement of LPRM strings provided SRM instrumentation is OPERABLE per Specification 3.10.B.

TABLE 3.1-1 (Continued)

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENTS

TABLE NOTATIONS

(a) CHANNEL may be placed in an inoperable status for up to 6 hours for required surveillance without placing the TRIP SYSTEM in the tripped condition provided at least one OPERABLE CHANNEL in the same TRIP SYSTEM is monitoring that parameter.

(b) Two TRIP SYSTEMS shall be OPERABLE in the applicable OPERATIONAL MODES for the specified Trip Function. CHANNEL OPERABILITY requirements within the TRIP SYSTEM are specified in the ACTION provisions of Specification 3.1.A.

(c) This function shall be automatically bypassed when the reactor mode switch is in the RUN position.

(d) The +8-inch trip point is the water level as measured by the instrumentation outside the shroud. The water level inside the shroud will decrease as power is increased to 100% in comparison to the level outside the shroud to a maximum of 7 inches. This is due to the pressure drop across the steam dryer. Therefore, at 100% power, an indication of +8 inch water level will actually be +1 inch inside the shroud. 1 inch on the water level instrumentation is ≥ 504" above vessel zero.

(e) CHANNEL shared by the reactor protection and containment isolation system.

(f) An APRM will be considered inoperable if there are fewer than 2 LPRM inputs per level or there are less than 50% of the normal complement of LPRMs to an APRM.

(g) This function is not required to be OPERABLE when the reactor pressure vessel head is not bolted to the vessel.

(h) This function is not required to be OPERABLE when PRIMARY CONTAINMENT INTEGRITY is not required.

(i) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.10.D or 3.10.E.

(j) Permissible to bypass when turbine first stage pressure is less than that which corresponds to 45% of rated steam flow (< 400 psi).

TABLE 3.1-1 (Continued)

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENTS

TABLE NOTATIONS

(k) The design permits closure of any one line without a scram being initiated.

(l) Permissible to bypass, with control rod block, for reactor protection system reset in REFUEL and SHUTDOWN positions of the reactor mode switch.

(m) The "shorting links" shall be removed from the RPS circuitry prior to and during the time any control rod is withdrawn and shutdown margin demonstrations are being performed. Not required for control rods removed per Specification 3.10.D or 3.10.E.

(n) The non-coincident NMS reactor trip function logic is such that all channels go to both trip systems. Therefore, when the "shorting links" are removed, the Minimum OPERABLE CHANNELS Per TRIP SYSTEM is 4 APRMS and 6 IRMS.

(o) Trip is indicative of turbine control valve fast closure (due to low EHC fluid pressure) as a result of fast acting valve actuation.

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TABLE 4.1-1

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION

CHANNEL FUNCTIONAL TEST AND CHANNEL CHECK REQUIREMENTS

Trip Function	CHANNEL FUNCTIONAL TEST Method (a)	CHANNEL FUNCTIONAL TEST (c)	CHANNEL CHECKS	Applicable OPERATIONAL MODES
1. Mode Switch in Shutdown	Place Mode Sw in Shutdown	R	NA	1, 2, 3, 4, 5
2. Manual Scram	Trip Channel and Alarm	W	NA	1, 2, 3, 4, 5
3. IRM				
a. High Flux	Trip Channel and Alarm (d)	S/U(e), W W	S/U, S, (b) S	2(k) 3, 4, 5
b. Inoperative	Trip Channel and Alarm	W	NA	2(k), 3, 4, 5
4. APRM				
a. High Flux (flow biased)	Trip Output Relays (d)	W(i), Q	S, D(l)	1
b. Inoperative	Trip Output Relays	Q	NA	1, 2, 3, 5
c. High Flux (15% scram)	Trip Output Relays (d)	S/U(e), W W	S/U, S, (b) S	2(k) 3, 5
d. High Flux (Scram Clamp)	Trip Output Relays (d)	W(i), Q	S	1
5. Reactor High Pressure	Trip Channel and Alarm	Q	NA	1, 2(h)
6. Drywell High Pressure	Trip Channel and Alarm	Q	NA	1, 2(j)
7. Reactor Low Water Level	Trip Channel and Alarm	Q	D	1, 2

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TABLE 4.1-1

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION

CHANNEL FUNCTIONAL TEST AND CHANNEL CHECK REQUIREMENTS

Trip Function	CHANNEL FUNCTIONAL TEST Methods (a)	CHANNEL FUNCTIONAL TEST (c)	CHANNEL CHECK	Applicable OPERATIONAL MODES
8. Scram Discharge Volume High Water Level (Thermal and dp Switch)	Trip Channel and Alarm (f)	Q	NA	1, 2, 5(g)
9. Turbine Condenser Low Vacuum	Trip Channel and Alarm	Q	NA	1
10. Main Steam Line High Radiation	Trip Channel and Alarm (d)	Q	S	1, 2(h)
11. Main Steam Line Isolation Valve Closure	Trip Channel and Alarm	Q	NA	1
12. Turbine Control Valve Fast Closure	Trip Channel and Alarm	Q	NA	1
13. Turbine Stop Valve Closure	Trip Channel and Alarm	Q	NA	1
14. Turbine EHC Control Fluid Low Pressure	Trip Channel and Alarm	Q	NA	1

TABLE 4.1-1 (Continued)

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION

CHANNEL FUNCTIONAL TEST AND CHANNEL CHECK REQUIREMENTS

TABLE NOTATIONS

(a) A CHANNEL 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.

(b) The IRM and SRM channels shall be determined to overlap for at least (1/2) decades during each startup after entering OPERATIONAL MODE 2 and the IRM and APRM channels shall be determined to overlap for at least (1/2) decades during each controlled shutdown, if not performed within the previous 7 days.

(c) CHANNEL 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.

(d) This instrumentation is exempted from the CHANNEL FUNCTIONAL TEST definition (Definition 1.6). This CHANNEL FUNCTIONAL TEST will consist of injecting a simulated electrical signal into the measurement CHANNELS.

(e) Within 24 hours prior to startup, if not performed within the previous 7 days.

(f) Only the electronics portion of the thermal switches will be tested using an electronic calibrator during the three month test. A water column or equivalent will be used to test the dp switches.

(g) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.10.D or 3.10.E.

(h) This function is not required to be OPERABLE when the reactor pressure vessel head is not bolted to the vessel.

(i) Within one week after entering OPERATIONAL MODE 1 and then quarterly thereafter.

(j) This function is not required to be OPERABLE when PRIMARY CONTAINMENT INTEGRITY is not required.

TABLE 4.1-1 (Continued)

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION

CHANNEL FUNCTIONAL TEST AND CHANNEL CHECK REQUIREMENTS

TABLE NOTATIONS

(k) The provisions of Specification 4.0.D are not applicable provided the CHANNEL FUNCTIONAL TEST is performed within 12 hours after entering OPERATIONAL MODE 2 from OPERATIONAL MODE 1.

(l) Verify measured core flow to be greater than or equal to established core flow at the existing pump speed.

TABLE 4.1-2

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION

CHANNEL CALIBRATION REQUIREMENTS

Trip Function	CHANNEL CALIBRATION Method (a) (f)	Minimum Frequency (b)	Applicable OPERATIONAL MODES
1. IRM High Flux	Electronic Calibration	R R	2 3, 4, 5
2. APRM High Flux			
a. Flow Bias	Standard Pressure and Voltage Source	W(d)(k), R	1
b. 15% Scram	Electronic Calibration	R R	2 3, 5
c. Scram Clamp	Electronic Calibration	W(d), R	1
3. LPRM (h)	Using TIP System	(g)	1
4. Reactor High Pressure	Standard Pressure Source	R	1, 2(j)
5. Drywell High Pressure	Standard Pressure Source	R	1, 2(i)
6. Reactor Low Water Level	Standard Pressure Source	R(e)	1, 2
7. Turbine Condenser Low Vacuum	Standard Vacuum Source	R	1
8. Main Steam Line High Radiation	Appropriate Radiation Source (c)	R	1, 2(j)
9. Turbine EHC Control Fluid Low Pressure	Standard Pressure Source	R	1
10. Turbine Control Valve Fast Closure	Standard Pressure Source	R	1
11. High Water Level in Scram Discharge Volume (dp only)	Standard Pressure Source	R	1, 2, 5

TABLE 4.1-2 (Continued)

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENTS

CHANNEL CALIBRATION REQUIREMENTS

TABLE NOTATIONS

(a) Neutron detectors may be excluded from the CHANNEL CALIBRATION.

(b) CHANNEL 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.

(c) A current source provides an instrument CHANNEL alignment every 3 months.

(d) This calibration shall consist of the adjustment of the APRM channel to conform to the power values calculated by a heat balance during OPERATIONAL MODE 1 when thermal power > 25% of RATED THERMAL POWER. Adjust the APRM channel if the absolute difference is greater than 2% of RATED THERMAL POWER. Any APRM channel gain adjustment made in compliance with Specification 2.1.A or 2.1.B shall not be included in determining the absolute difference.

(e) Trip units are calibrated at least once per quarter and transmitters are calibrated at least once per OPERATING CYCLE.

(f) Response time is not part of the routine CHANNEL CHECK and CHANNEL CALIBRATION.

(g) Every 1000 equivalent full power hours.

(h) Does not provide scram function.

(i) This function is not required to be OPERABLE when PRIMARY CONTAINMENT INTEGRITY is not required.

(j) This function is not required to be OPERABLE when the reactor pressure vessel head is not bolted to the vessel.

(k) This Calibration shall consist of the adjustment of the APRM flow biased channel to conform to a calibrated flow signal.

3.1 LIMITING CONDITIONS FOR OPERATION BASES

The reactor protection system automatically initiates a reactor scram to:

a. preserve the integrity of the fuel cladding,

b. preserve the integrity of the primary system, and

c. 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. When necessary, one channel may be made inoperable for brief intervals to conduct required functional tests and calibrations.

The reactor protection system is of the dual channel type (reference SAR Section 7.7.1.2). The system is made up of two independent trip systems, each having two subchannels of tripping devices. Each subchannel has an input from at least one instrument channel which monitors a critical parameter.

The outputs of the subchannels are combined in a one-out-of-two-logic, i.e., an input signal on either one or both of the subchannels will cause a trip system trip. The outputs of the trip systems are arranged so that a trip on both systems is required to produce a reactor scram.

This system meets the requirements of the IEEE 279, "Standard for Nuclear Power Plant Protection Systems" issued September 13, 1966. The system has a reliability greater than that of a two-out-of-three system and somewhat less than that of a one-out-of-two system (reference APED 5179).

With the exception of the average power range monitor (APRM) and intermediate range monitor (IRM) channels, each subchannel has at least one instrument channel. When the minimum condition for operation on the number of operable instrument channels per untripped protection trip system is met, or if it cannot be met and the affected protection trip system is placed in a tripped condition, 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.

APRMs #1 and #3 operate contacts in one subchannel and APRMs #2 and #3 operate contacts in the other subchannel. APRMs #4, #5 and #6 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 per channel. This allows the bypassing of one APRM per protection trip system for maintenance, testing, or calibration. Additional IRM channels have also been provided to allow for bypassing of one such channel. 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 Specifications 2.1 and 2.2.

Pressure sensing of the drywell is provided to detect a loss-of-coolant accident and initiate the emergency core cooling equipment. The pressure-sensing instrumentation is a backup to the water-level instrumentation which is discussed in Specification 2.1. A scram is provided at the same setting as the emergency core cooling system (ECCS) initiation to minimize the energy which must be accommodated during a loss-of-coolant accident and to prevent the reactor from going critical following the accident.

The control rod drive scram system is designed so that all of the water which is discharged from the reactor by a scram can be accommodated in the discharge piping. A part of this system is an individual instrument volume for each of the south and north CRD accumulators. These two volumes and their piping can hold in excess of 90 gallons of water and are the low point in the piping. No credit was taken for these volumes in the design of the discharge piping relative to the amount of water which must be accommodated during a scram. During normal operations, the discharge volumes are empty; however, should either volume fill with water, the water discharged to the piping from the reactor may not be accommodated which could result in slow scram times or partial or no control rod insertion. To preclude this occurrence, level switches have been installed in both volumes which will alarm and scram the reactor when the volume remaining in either instrument volume is approximately 40 gallons. For diversity of level sensing methods that will ensure and provide a scram, both differential pressure switches and thermal switches have been incorporated into the design and logic of the system. The setpoint for the scram signal has been chosen on the basis of providing sufficient volume remaining to accommodate a scram, even with 5 gpm leakage per drive into the SDV. As indicated above, there is sufficient volume in the piping to accommodate the scram without impairment of the scram times or the 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 properly.

Loss of condenser vacuum occurs when the condenser can no longer handle 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 cladding 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 cladding safety limit from being exceeded in the event of a turbine trip transient with bypass closure.

The condenser low-vacuum scram is a backup to the stop valve closure scram and causes a scram before the stop valves are closed, thus the resulting transient is less severe. Scram occurs at 21-inches Hg vacuum, stop valve closure occurs at 20-inches Hg vacuum, and bypass closure at 7-inches Hg vacuum.

High radiation levels in the main steamline tunnel above that due to the normal nitrogen and oxygen radioactivity are an indication of leaking fuel. A scram is initiated whenever such radiation level exceeds fifteen times normal background (without hydrogen addition). The purpose of this scram is to reduce the source of such radiation to the extent necessary to prevent excessive turbine contamination. Discharge of excessive amounts of radioactivity to the site environs is prevented by the air ejector off-gas monitors, which cause an isolation of the main condenser off-gas line provided the limit specified in Specification 3.8 is exceeded.

The main steamline isolation valve closure scram is set to scram when the isolation valves are 10% closed from full open. 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 (reference SAR Section 7.7.1.2). Whenever the reactor mode switch is in the REFUEL or STARTUP HOT STANDBY position, the turbine condenser low-vacuum scram and main steamline isolation valve closure scrams are bypassed. This bypass has been provided for flexibility

during startup and to allow repairs to be made to the turbine condenser. While this bypass is in effect, protection is provided against pressure or flux increases by the high-pressure scram and APRM 15% scram, respectively, which are effective in STARTUP/HOT STANDBY.

If the reactor was taken to a hot standby condition for repairs to the turbine condenser, the main steamline isolation valves would be closed. No hypothesized single failure or single operator action in this mode of operation can result in an unreviewed radiological release.

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

The IRM system provides protection against excessive power levels and short reactor periods in the startup and intermediate power ranges (reference SAR Sections 7.4.4.2 and 7.4.4.3). During refueling, the primary Neutron Monitoring System (NMS) indication of neutron flux levels is provided by the Source Range Monitors (SRM). The SRMs provide input to the RPS, but shorting links are installed across the normally closed contacts such that tripping an SRM channel does not affect the RPS. To activate the SRM scram function, these shorting links must be removed from the RPS. The SRM control rod scram provides backup protection to refueling interlocks and SHUTDOWN MARGIN should a prompt reactivity excursion occur. Although the IRM and APRM functions are required to be OPERABLE during refueling, the SRMs provide the only on-scale monitoring of neutron flux levels during refueling and therefore the shorting links must be removed to enable the scram function of the SRMs. The RPS (and therefore removal of the RPS shorting links) is required to be OPERABLE in REFUEL only with any control rod withdrawn from a core cell containing one or more fuel assemblies. Control rods withdrawn from a core cell containing no fuel assemblies do not affect the reactivity of the core and therefore are not required to have the capability to scram. Provided all control rods are otherwise inserted, the RPS function is not required. In this condition, the required SHUTDOWN MARGIN and the one-rod-out interlock provide assurance that the reactor will not become critical thereby requiring a scram. If the SHUTDOWN MARGIN has been demonstrated, the RPS shorting links are not required to be removed. Under these conditions, the capability of the one-rod-out interlock to prevent criticality has been demonstrated and the backup scram protection provided by the IRMs is sufficient to ensure a highly reliable scram if required. In the power range, the APRM system provides

required protection (reference SAR Section 7.4.5.2). Thus, the IRM system is not required in the RUN OPERATIONAL MODE, the APRMs cover only the intermediate and power range; and the IRMs provide adequate coverage in the startup and intermediate range.

The high reactor pressure, high drywell pressure, low reactor water level scrams are required for OPERATIONAL MODES 1 and 2. The scram discharge volume high level scram is required in OPERATIONAL MODES 1, 2 and 5. They are therefore required to be operational for these OPERATIONAL MODES of reactor operation.

The turbine condenser low-vacuum scram is required only during power operation and must be bypassed to start up the unit.

4.1 SURVEILLANCE REQUIREMENTS BASES

A. Surveillance requirements for the reactor protection system are selected in order to demonstrate proper function and operability. The surveillance intervals are determined in many different ways, such as, 1) operating experience, 2) good engineering judgement, 3) reliability analyses, or 4) other analyses that are found acceptable to the NRC.

The frequency of calibration of the APRM flow-biasing network has been established at each refueling outage. The flow-biasing network is functionally tested at least once per quarter and, in addition, cross calibration checks of the flow input to the flow-biasing network can be made during the functional test by direct meter reading (IEEE 279 Standard for Nuclear Power Plant Protection Systems, Section 4.9, September 13, 1966). There are several instruments which must be calibrated, and will take several days to perform the calibration of the entire network. While the calibration is being performed, a zero flow signal will be sent to half of the APRMs, resulting in a half scram and rod block condition. Thus, if the calibrations were performed during operation, flux shaping would not be possible. Based on experience at other generating stations, drift of instruments such as those in the flow-biasing network, is not significant; therefore, to avoid spurious scrams, a calibration frequency of each refueling outage is established.

Reactor low water level instruments 1(2)-263-57A, 1(2)-263-57B, 1(2)-263-58A, and 1(2)-263-58B have been modified to be an analog trip system. The analog trip system consists of an analog sensor (transmitter) and a master/slave trip unit setup which ultimately drives a trip relay. The frequency of calibration for the trip unit has been established in General Electric topical Report NEDC-30851P-A as quarterly. An adequate calibration/ surveillance test interval for the transmitter is once per operating cycle.

Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with NEDC-30851P-A, "Technical Specification Improvement Analysis for BWR Reactor Protection System," as approved by the NRC in a letter dated July 15, 1987 from A. Thadani to T.A. Pickens .

The turbine control valve fast acting solenoid valve pressure switches directly measure the trip oil pressure that causes the turbine control valves to close in a rapid manner. The reactor scram setpoint was developed in accordance with NEDC-31336 "General Electric Instrument Setpoint Methodology" dated October, 1986. As part of the calculation, a calibration period is inputted to achieve a nominal trip point and an allowable setpoint (Technical Specification value). The nominal setpoint is procedurally controlled. Based on the calculation input, the calibration period is defined to be every Refueling Outage.

The sensitivity of LPRM detectors decreases with exposure to neutron flux at a slow and approximately constant rate. Changes in power distribution and electronic drift also require compensation. This compensation is accomplished by calibrating the APRM system every 7 days using heat balance data and by calibrating individual LPRM's at least every 1000 equivalent full-power hours using TIP traverse data. Calibration on this frequency assures plant operation at or below thermal limits.

A comparison of Tables 4.1-1 and 4.1-2 indicates that some instrument channels have not been included in the latter table. These are mode switch in shutdown, manual scram, main steamline isolation valve closure, and turbine stop valve closure. All of the devices or sensors associated with these scram functions are simple on-off switches, hence calibration is not applicable, i.e., the switch is either on or off. Further, these switches are mounted solidly to the device and have a very low probability of moving; e.g., the thermal switches in the scram discharge volume tank. Based on the above, no calibration is required for these instrument channels.

B. The MFLPD shall be checked once per day to determine if the APRM scram requires adjustment. This may normally be done by checking the LPRM readings, TIP traces, or process computer calculations. Only a small number of control rods are moved daily, thus the peaking factors are not expected to change significantly and a daily check of the MFLPD is adequate.

References

1. Licensing Topical Report NEDO-21617-A (December 1978).
2. General Electric Topical Report NEDC-30851P-A.
3. NEDC-31336 "General Electric Instrument Setpoint Methodology" dated October, 1986.

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3.1/4.1 REACTOR PROTECTION SYSTEM

LIMITING CONDITIONS FOR OPERATION

Applicability:

Applies to instrumentation and associated devices which initiate a reactor scram.

Objective:

To assure the operability of the reactor protection system.

SURVEILLANCE REQUIREMENTS

Applicability:

Applies to the surveillance of the instrumentation and associated devices which initiate reactor scram.

Objective:

To specify the type and frequency of surveillance to be applied to the protection instrumentation.

SPECIFICATIONS

A. The setpoints, minimum number of trip systems, and minimum number of instrument channels that must be operable for each position of the reactor mode switch shall be as given in Tables 3.1-1 through 3.1-4. The system response times from the opening of the sensor contact up to and including the opening of the trip actuator contacts shall not exceed 50 milliseconds.

B. If, during operation, the maximum fraction of limiting power density exceeds the fraction of rated power when operating above 25% rated thermal power, either:

1. The APRM scram and rod block settings shall be reduced to the values given by the equations in Specification 2.1.A.1 and 2.1.B. This may also be accomplished by increasing the APRM gain as described therein.

A. Instrumentation systems shall be functionally tested and calibrated as indicated in Tables 4.1-1 and 4.1-2 respectively.

B. Daily during reactor power operation, the core power distribution shall be checked for maximum fraction of limiting power density (MFLPD) and compared with the fraction of rated power (FRP) when operating above 25% rated thermal power.

2. The power distribution shall be changed such that the maximum fraction of limiting power density no longer exceeds the fraction of rated power.

C. When it is determined that a channel is failed in the unsafe condition and Column 1 of Tables 3.1-1 through 3.1-3 cannot be met, that trip system must be put in the tripped condition immediately. All other RPS channels that monitor the same variable shall be functionally tested within 8 hours. The trip system with the failed channel may be untripped for a period of time not to exceed 1 hour to conduct this testing. As long as the trip system with the failed channel contains at least one operable channel monitoring that same variable, that trip system may be placed in the untripped position for short periods of time to allow functional testing of all RPS instrument channels as specified by Table 4.1-1. The trip system may be in the untripped position for no more than 8 hours per functional test period for this testing.

3.1 LIMITING CONDITIONS FOR OPERATION BASES

The reactor protection system automatically initiates a reactor scram to:

a. preserve the integrity of the fuel cladding,

b. preserve the integrity of the primary system, and

c. 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. When necessary, one channel may be made inoperable for brief intervals to conduct required functional tests and calibrations.

The reactor protection system is of the dual channel type (reference SAR Section 7.7.1.2). The system is made up of two independent trip systems, each having two subchannels of tripping devices. Each subchannel has an input from at least one instrument channel which monitors a critical parameter.

The outputs of the subchannels are combined in a one-out-of-two logic, i.e., an input signal on either one or both of the subchannels will cause a trip system trip. The outputs of the trip systems are arranged so that a trip on both systems is required to produce a reactor scram.

This system meets the requirements of the IEEE 279, Standard for Nuclear Power Plant Protection Systems issued September 13, 1966. The system has a reliability greater than that of a two-out-of-three system and somewhat less than that of a one-out-of-two system (reference APED 5179).

With the exception of the average power range monitor (APRM) and intermediate range monitor (IRM) channels, each subchannel has at least one instrument channel. When the minimum condition for operation on the number of operable instrument channels per untripped protection trip system is met, or if it cannot be met and the affected protection trip system is placed in a tripped condition, 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.

APRM's # 1 and # 3 operate contacts in one subchannel and APRM's # 2 and #3 operate contacts in the other subchannel. APRM's # 4, # 5 and # 6 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 per channel. This allows the bypassing of one APRM per protection trip system for maintenance, testing, or calibration. Additional IRM channels have also been provided to allow for bypassing of one such channel. 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 Specifications 2.1 and 2.2.

Pressure sensing of the drywell is provided to detect a loss-of-coolant accident and initiate the emergency core cooling equipment. The pressure-sensing instrumentation is a backup to the water-level instrumentation which is discussed in Specification 2.1. A scram is provided at the same setting as the emergency core cooling system (ECCS) initiation to minimize the energy which must be accommodated during a loss-of-coolant accident and to prevent the reactor from going critical following the accident.

The control rod drive scram system is designed so that all of the water which is discharged from the reactor by a scram can be accommodated in the discharge piping. A part of this system is an individual instrument volume for each of the south and north CRD accumulators. These two volumes and their piping can hold in excess of 90 gallons of water and ~~is~~ are the low point in the piping. No credit was taken for these volumes in the design of the discharge piping relative to the amount of water which must be accommodated during a scram. During normal operations, the discharge volumes are empty; however, should either volume fill with water, the water discharged to the piping from the reactor may not be accommodated which could result in slow scram times or partial or no control rod insertion. To preclude this occurrence, level switches have been installed in both volumes which will alarm and scram the reactor when the volume remaining in either instrument volume is approximately 40 gallons. For diversity of level sensing methods that will ensure and provide a scram, both differential pressure switches and thermal switches have been incorporated into the design and logic of the system. The setpoint for the scram signal has been chosen on the basis of providing sufficient volume remaining to accommodate a scram, even with 5 gpm leakage per drive into the SDV. As indicated above, there is sufficient volume in the piping to accommodate the scram without impairment of the scram times or the 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 properly.

Loss of condenser vacuum occurs when the condenser can no longer handle 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 cladding 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 cladding safety limit from being exceeded in the event of a turbine trip transient with bypass closure.

The condenser low-vacuum scram is a backup to the stop valve closure scram and causes a scram before the stop valves are closed, thus the resulting transient is less severe. Scram occurs at 21 inches Hg vacuum, stop valve closure occurs at 20 inches Hg vacuum, and bypass closure at 7 inches Hg vacuum.

High radiation levels in the main steamline tunnel above that due to the normal nitrogen and oxygen radioactivity are an indication of leaking fuel. A scram is initiated whenever such radiation level exceeds fifteen times normal background (without hydrogen addition). The purpose of this scram is to reduce the source of such radiation to the extent necessary to prevent excessive turbine contamination. Discharge of excessive amounts of radioactivity to the site environs is prevented by the air ejector off-gas monitors, which cause an isolation of the main condenser off-gas line provided the limit specified in Specification 3.8 is exceeded.

The main steamline isolation valve closure scram is set to scram when the isolation valves are 10% closed from full open. 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 (reference SAR Section 7.7.1.2). Whenever the reactor mode switch is in the REFUEL or STARTUP/HOT STANDBY position, the turbine condenser low-vacuum scram and main steamline isolation valve closure scram are bypassed. This bypass has been provided for flexibility during startup and to allow repairs to be made to the turbine condenser. While this bypass is in effect, protection is provided against pressure or flux increases by the high-pressure scram and APRM 15% scram, respectively, which are effective in ~~this mode~~. STARTUP/HOT STANDBY.

If the reactor was taken ~~were brought~~ to a hot standby condition for repairs to the turbine condenser, the main steamline isolation valves would be closed. No hypothesized single failure or single operator action in this mode of operation can result in an unreviewed radiological release.

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

The IRM system provides protection against excessive power levels and short reactor periods in the startup and intermediate power ranges (reference SAR Sections 7.4.4.2 and 7.4.4.3). ~~A source range monitor (SRM) system is also provided to supply additional~~

insert 'A' ~~neutron level information during startup but has no scram functions (reference SAR Section 7.4.3.2). Thus the IRM is required in the REFUEL and STARTUP/HOT STANDBY modes. In addition, protection is provided in this range by the APRM 15% scram as discussed in the bases for Specification 2.1.~~ In the power range, the APRM system provides required protection (reference SAR Section 7.4.5.2). Thus, the IRM system is not required in the RUN OPERATIONAL MODE, the APRM's cover only the intermediate and power range; and the IRM's provide adequate coverage in the startup and intermediate range.

The high reactor pressure, high drywell pressure, reactor low water level, ~~and scram discharge volume high level scrams are required for the Start/Hot Standby and Run modes of plant operation.~~ scrams are required for OPERATIONAL MODES 1 and 2. The scram discharge volume high level scram is required in OPERATIONAL MODES 1, 2 and 5. They are therefore required to be operational for these OPERATIONAL ~~modes~~ of reactor operation.

The turbine condenser low-vacuum scram is required only during power operation and must be bypassed to start up the unit.

~~The requirement that the IRM's be inserted in the core when the APRM's read 3/125 of full scale assures that there is proper overlap in the neutron-monitoring systems and thus that adequate coverage is provided for all ranges of reactor operation.~~

4.1 SURVEILLANCE REQUIREMENTS BASES

A. The minimum functional testing frequency used in this specification is based on a reliability analysis using the concepts developed in Reference 1. This concept was specifically adapted to the one-out-of-two taken twice logic of the reactor protection system. The analysis shows that the sensors are primarily responsible for the reliability of the reactor protection system.. This analysis makes use of "unsafe failure" rate experience at conventional and nuclear power plants in a reliability model for the system. An unsafe failure is defined as one which negates channel operability and which, due to its nature, is revealed only when the channel is functionally tested or attempts to respond to a real signal. Failures such as blown fuses, ruptured bourdon tubes, faulted amplifiers, faulted cables, etc., which result in "upscale" or "downscale" readings on the reactor instrumentation are "safe" and will be easily recognized by the operators during operation because they are revealed by an alarm or a scram.

The channels listed in Table 4.1-1 and 4.1-2 are divided into three groups respecting functional testing.

These are:

1. On-off sensors that provide a scram trip function (Group 1);
2. Analog devices coupled with bistable trips that provide a scram function (Group 2); and,
3. Devices which serve a useful function only during some restricted mode of operation, such as Startup/Hot Standby, Refuel, or Shutdown, or for which the only practical test is one that can be performed at shutdown (Group 3).

The sensors that make up Group 1 are specifically selected from among the whole family of industrial on-off sensors that have earned an excellent reputation for reliable operation. Actual history on this class of sensors operating in nuclear power plants shows four failures in 472 sensor years, or a failure rate of 0.97 X 10^{-6}/hr. During design a goal of 0.99999 probability of success (at the 50% confidence level) was adopted to assure that a balanced and adequate design is achieved. The probability of success is primarily a function of the sensor failure rate and the test interval. A 3-month test interval was planned for Group 1 sensors. This is in keeping with good operating practice and satisfies the design goal for the logic configuration utilized in the reactor protection system.

[Insert "B"]

To satisfy the long-term objective of maintaining an adequate level of safety throughout the plant lifetime, a minimum goal of 0.9999 at the 95% confidence level is proposed. With the one-out-of-two taken twice logic, this requires that each sensor have an availability of 0.993 at the 95% confidence level. This level of availability may be maintained by adjusting the test interval as a function of the observed failure history (Reference 1). To facilitate the implementation of this technique, Figure 4.1-1 is provided to indicate an appropriate trend in test interval. The procedure is as follows:

1. Like sensors are pooled into one group for the purpose of data acquisition.
2. The factor M is the exposure hours and is equal to the number of sensors in a group, n, times the elapsed time T(M=nT).
3. The accumulated number of unsafe failures is plotted as an ordinate against M as an abscissa on Figure 4.1-1.
4. After a trend is established, the appropriate monthly test interval to satisfy the goal will be the test interval to the left of the plotted points.
5. A test interval of 1 month will be used initially until a trend is established.

MOVE TO AFTER LAST PARAGRAPH ON NEXT PAGE

The turbine control valve fast acting solenoid valve pressure switches directly measure the trip oil pressure that causes the turbine control valves to close in a rapid manner. The reactor scram setpoint was developed in accordance with NEDC 31336 "General Electric Instrument Setpoint Methodology" dated October, 1986. As part of the calculation, a calibration period is inputted to achieve a nominal trip point and an allowable setpoint (Technical Specification value). The nominal setpoint is procedurally controlled. Based on the calculation input, the calibration period is defined to be every Refueling Outage.

Group 2 devices utilize an analog sensor followed by an amplifier and a bistable trip circuit. The sensor and amplifier are active components, and a failure is almost always accompanied by an alarm and an indication of the source of trouble. In the event of failure, repair or substitution can start immediately. An as-is failure is one that "sticks" midscale and is not capable of going either up or down in response to an out-of-limits input. This type of failure for analog devices is a rare occurrence and is detectable by an operator who observes that one signal does not track the other three. For purposes of analysis, it is assumed that this rare failure will be detected within 2 hours.

The bistable trip circuit which is a part of the Group 2 devices can sustain unsafe failures which are revealed only on test. Therefore, it is necessary to test them periodically.

A study was conducted of the instrumentation channels included in the Group 2 devices to calculate their 'unsafe' failure rates. The analog devices (sensors and amplifiers) are predicted to have an unsafe failure rate of less than 20×10^{-6} failures/hour. The bistable trip circuits are predicted to have an unsafe failure rate of less than 2×10^{-6} failures/hours. Considering the 2-hour monitoring interval for the analog devices as assumed above and a weekly test interval for the bistable trip circuits, the design reliability goal of 0.99999 is attained with ample margin.

~~The bistable devices are monitored during plant operation to record their failure history and establish a test interval using the curve of Figure 4.1-1. There are numerous identical bistable devices used throughout the plant instrumentation system. Therefore, significant data on the failure rates for the bistable devices should be accumulated rapidly.~~

quarter

The frequency of calibration of the APRM flow-biasing network has been established at each ~~refueling outage~~. The flow-biasing network is functionally tested at least once per ~~month~~ and, in addition, cross calibration checks of the flow input to the flow-biasing network can be made during the functional test by direct meter reading (IEEE 279 Standard for Nuclear Power Plant Protection Systems, Section 4.9, September 13, 1966). There are several instruments which must be calibrated, and it will take several days to perform the calibration of the entire network. While the calibration is being performed, a zero flow signal will be sent to half of the APRM's, resulting in a half scram and rod block condition. Thus, if the calibration were performed during operation, flux shaping would not be possible. Based on experience at other generating stations, drift of instruments, such as those in the flow-biasing network, is not significant; therefore, to avoid spurious scrams, a calibration frequency of each refueling outage is established. (5)

Reactor low water level instruments (2) 1-263-57A, (2) 1-263-57B, (2) 1-263-58A, and (2) 1-263-58B have been modified to be an analog trip system. The analog trip system consists of an analog sensor (transmitter) and a master/slave trip unit setup which ultimately drives a trip relay. The frequency of calibration ~~and functional testing for instrument loops of the analog trip system, including reactor low water level, has been established in Licensing Topical Report NEDO-21617-A (December 1978). With the one-out-of-two-taken-twice logic, NEDO-21617-A states that each trip unit be subjected to a calibration/functional test of one month~~. An adequate calibration/surveillance test interval for the transmitter is once per operating cycle.

specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with NEDC-30851P-A, "Technical Specification Improvement Analyses for BWR Reactor Protection System", as approved by the NRC in a letter dated July 15, 1987 from A. Thadani to T. A. Pickens
[Add paragraph from page 3.1/4.1-8]

for the trip unit has been established in General Electric Topical Report NEDC-30851P-A as quarterly.

~~Group 3 devices are active only during a given portion of the operation cycle. For example, the IRM is active during startup and inactive during full-power operation. Thus, the only test that is meaningful is the one performed just prior to shutdown or startup, i.e., the tests that are performed just prior to use of the instrument.~~

~~Calibration frequency of the instrument channel is divided into two groups. These are as follows:~~

1. ~~Passive type indicating devices that can be compared with like units on a continuous basis, and~~

2. ~~Vacuum tube or semiconductor devices and detectors that drift or lose sensitivity.~~

~~Experience with passive type instruments in Commonwealth Edison generating stations and substations indicate that the specified calibrations are adequate. For those devices which employ amplifiers, etc. drift specifications call for drift to be less than 0.4%/month i.e., in the period of a month a drift of 0.4% would occur, thus providing for adequate margin.~~

The sensitivity of LPRM detectors decreases with exposure to neutron flux at a slow and approximately constant rate. Changes in a power distribution and electronic drift also require compensation. This compensation is accomplished by calibrating the APRM system every 7 days using heat balance data by calibrating individual LPRM's at least every 1000 equivalent full-power hours using TIP traverse data. Calibration on this frequency assures plant operation at or below thermal limits.

A comparison of Tables 4.1-1 and 4.1-2 indicates that some instrument channels have not been included in the latter table. These are mode switch in shutdown, manual scram, high water level in scram discharge volume, main steamline isolation valve closure, and turbine stop valve closure. All of the devices or sensors associated with these scram functions are simple on-off switches, hence calibration is not applicable, i.e., the switch is either on or off. Further, these switches are mounted solidly to the device and have a very low probability of moving; e.g., the thermal switches in the scram discharge volume tank. Based on the above, no calibration is required for these instrument channels.

B. The MFLPD shall be checked once per day to determine if the APRM scram requires adjustment. This may normally be done by checking the LPRM readings, TIP traces, or process computer calculations. Only a small number of control rods are moved daily, thus the peaking factors are not expected to change significantly and a daily check of the MFLPD is adequate.

References

~~1. I. M. Jacobs, "Reliability of Engineered Safety Features as a Function of Testing Frequency", Nuclear Safety, Vol. 9, No. 4, pp. 310-312, July-August 1968.~~

1 ~~2.~~ Licensing Topical Report NEDO-21617-A (December 1978).

3. NEDC - 31336 "General Electric Instrument Setpoint Methodology" dated October, 1986

2. General Electric Topical Report NEDC-30851P-A

INSERT FOR TECHNICAL SPECIFICATION
SECTION 3.1/4.1 "REACTOR PROTECTION SYSTEM"

Insert "A"

During refueling, the primary Neutron Monitoring System (NMS) indication of neutron flux levels is provided by the Source Range Monitors (SRM). The SRMs provide input to the RPS, but shorting links are installed across the normally closed contacts such that tripping an SRM channel does not affect the RPS. To activate the SRM scram function these shorting links must be removed from the RPS. The SRM control rod scram provides backup protection to refueling interlocks and SHUTDOWN MARGIN should a prompt reactivity excursion occur. Although the IRM and APRM functions are required to be OPERABLE during refueling, the SRMs provide the only on-scale monitoring of neutron flux levels during refueling and therefore the shorting links must be removed to enable the scram function of the SRMs. The RPS (and therefore removal of the RPS shorting links) is required to be OPERABLE in REFUEL only with any control rod withdrawn from a core cell containing one or more fuel assemblies. Control rods withdrawn from a core cell containing no fuel assemblies do not affect the reactivity of the core and therefore are not required to have the capability to scram. Provided all control rods are otherwise inserted, the RPS function is not required. In this condition, the required SHUTDOWN MARGIN and the one-rod-out interlock provide assurance that the reactor will not become critical thereby requiring a scram. If the SHUTDOWN MARGIN has been demonstrated, the RPS shorting links are not required to be removed. Under these conditions, the capability of the one-rod-out interlock to prevent criticality has been demonstrated and the backup scram protection provided by the IRMs is sufficient to ensure a highly reliable scram if required.

Insert "B"

Surveillance requirements for the reactor protection system are selected in order to demonstrate proper function and operability. The surveillance intervals are determined in many different ways, such as, 1) operating experience, 2) good engineering judgement, 3) reliability analyses, or 4) other analyses that are found acceptable to the NRC.

SIGNIFICANT HAZARDS CONSIDERATIONS
AND
ENVIRONMENTAL ASSESSMENT EVALUATION

PROPOSED TS 3.1/4.1

"REACTOR PROTECTION SYSTEM"

EVALUATION FOR SIGNIFICANT HAZARDS CONSIDERATION

PROPOSED SPECIFICATION 3.1/4.1

REACTOR PROTECTION SYSTEM

The proposed changes provided in this amendment request are made in order to provide a more user friendly document, incorporate desired technical improvements, and to incorporate some improvements from later operating BWRs. These changes have been reviewed by Commonwealth Edison and we believe that they do not present a Significant Hazards Consideration. The basis for our determination is documented as follows:

BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION

Commonwealth Edison has evaluated this proposed amendment and determined that it involves no significant hazards consideration. In accordance with the criteria of 10 CFR 50.92(c) a proposed amendment to an operating license involves no significant hazards consideration if operation of the facility, in accordance with the proposed amendment, would not:

1) Involve a significant increase in the probability or consequences of an accident previously evaluated, because:

   a. The Generic Changes to the technical specifications involve administrative changes to format and arrangement of the material. As such, these changes cannot involve a significant increase in the probability or consequences of an accident previously evaluated.

   b. The proposed changes to Specifications 3.1.A/4.1.A and 3.1.B/4.1.B are made to provide the user with a format that will allow quicker access to needed information ar to provide concise LCO, Applicability, Action and Surveillance requirements. The blend of requirements from the present Quad Cities Technical Specifications and later operating BWRs utilizes proven material and testing techniques. The deletion of Surveillance Requirement 4.1.C on additional testing of RPS channels if one fails in the unsafe position does not significantly decrease the reliability of the RPS system. This additional testing may or may not find more problems in the system such as common mode failures. Evaluations to determine cause of the failure and the potential for additional failures in similar equipment provides an equivalent level of safety in the plant as the present testing requirements of 4.1.C.

      The proposed changes to Tables 3.1-1 through 3.1-3, 4.1-1 and 4.1-2 do not alter any established setpoints or reduce the minimum operable channels per trip system requirements. The proposed changes are applicable for the

Quad Cities plant and are current plant operating practice or have been utilized on other operating plants; therefore, they do not involve a significant increase in the probability or consequences of an accident previously evaluated.

c. The proposed changes to incorporate the Surveillance Testing Intervals and Allowed Out of Service Intervals in Topical Report NEDC-30851P-A do not degrade the reliability of the RPS system, as demonstrated in the Topical Report and corresponding plant specific analyses. Section 5.7.4 of NEDC-30851P-A provides a detailed generic Determination of No Significant Hazards for the proposed change. Implementation of the extended surveillance intervals for the Channel Functional Tests and Channel Calibrations will not be made without factoring in appropriate drift information into the setpoint calculations. Since the changes do not degrade the reliability of the RPS system over present conditions, there is no significant increase in the probability or consequences of an accident previously evaluated.

d. The proposed change to delete the APRM Downscale Scram Trip Function has been evaluated by Commonwealth Edison and General Electric. The accidents of concern with respect to the APRM/IRM companion trip are the Rod Drop Accident (RDA) and the low power Rod Withdrawal Error (RWE). FSAR and reload safety analyses do not credit this scram function in the termination of either of these accidents. Since this scram function is not credited in the termination of these accidents, the elimination of this scram function has no adverse effect of previously evaluated accidents.

2) Create the possibility of a new or different kind of accident from any previously evaluated because:

a. Since the Generic Changes proposed to the technical specifications are administrative in nature, they cannot create the possibility of a new or different kind of accident from any previously evaluated.

b. The changes to Specifications 3.1.A/4.1.A and 3.1.B/4.1.B blend STS requirements with existing Quad Cities requirements to provide a user friendly format and presentation of requirements. The deletion of Surveillance Requirement 4.1.C concerning additional testing of RPS channels if a channel fails in an unsafe position does not create the potential of a new or different kind of accident since other means are utilized to determine potential for common cause or similar failures in other channels. Many of the changes proposed to the Tables follow later operating BWR guidelines that are presently being utilized at these plants and have been

evaluated and found acceptable for use at Quad Cities. Other changes to the tables provide clarification of present requirements. Therefore, the changes do not create the possibility of a new or different kind of accident from any previously evaluated.

c. The proposed changes to incorporate the Surveillance Testing Intervals and Allowed Out of Service Times in Topical Report NEDC-30851P-A do not create the possibility of a new or different kind of accident from any previously evaluated because RPS function and reliability is not degraded by these changes. No new modes of plant operation are involved. The implementation of STS Channel Calibration Test frequencies will only be made to the extent that the instrumentation drift characteristics allow the interval extensions.

d. The deletion of the APRM Downscale Scram Trip Function does not introduce any new accident scenario. The limiting accidents (i.e., RDA and RWE) in the operating region of transition between the Startup and Run Operational Modes are well understood and are evaluated in FSAR and/or reload safety analyses. Other control rod initiated events which are less limiting in this region, such as fast period events (either due to operator error or CRD malfunction), are subsets of the low power RWE event and are bounded by it and the Design Basis RDA. General Electric has indicated that, for reactivity insertion mechanisms at very low power (if postulated to occur coincident with an inappropriate mode switch), the only effect of the deletion of the APRM downscale scram would be that the initial power level could be a few percent lower which would not have a significant effect on the severity of the event. In addition, proper overlap between the IRMs and APRMs is not affected since the calibration requirements are not being changed.

3) Involve a significant reduction in the margin of safety because:

a. The Generic Changes proposed in this amendment request are administrative in nature and, as such, do not involve a reduction in the margin of safety.

b. The changes to Specifications 3.1.A/4.1.A and 3.1.B/4.1.B implement an STS type of format while retaining the present two column layout. This two column layout has been in use at Quad Cities since initial licensing and is preferred by the majority of the technical specification users at the plant. The proposed LCO, Applicability, Actions and Surveillance Requirements are modeled after STS requirements which have been evaluated and found to be acceptable for use at Quad Cities. The deletion of present Surveillance Requirement 4.1.C does not involve a

significant reduction in the margin of safety since other equivalent methods are utilized to determine if the failure of one RPS channel in an unsafe position affects other similar channels.

The changes to the Tables in Section 3.1/4.1 follow proven STS guidelines that have been implemented at other operating BWR plants. These changes have been evaluated for use at Quad Cities with a determination that implementation at the plant will not involve a significant reduction in the margin of safety. Other changes to the tables involve clarifications or minor improvements that do not affect the margin of safety.

c. The changes proposed in Topical Report NEDC-30851P-A increases the testing for the Manual Scram function and decreases testing for the other applicable Scram functions. Allowed out of service times are increased for the RPS channels as a result of the Topical Report analyses. However, the requested changes do not degrade the reliability of the RPS system and thus the margin of safety is preserved. The results of the topical report have been found acceptable for plant use by NRC SER with the stipulation that setpoint drift over the increased testing interval be considered in setpoint calculations. Quad Cities will consider the additional drift in the setpoint calculations before implementing the extended surveillance testing intervals for both the Channel Functional Tests and the Channel Calibration Tests. Therefore, the changes do not involve a significant reduction in the margin of safety.

d. The APRM Downscale Scram Trip Function is not credited in the termination of any FSAR or reload safety analysis event. As such, the elimination of this scram function has no effect on any margin of safety.

ENVIRONMENTAL ASSESSMENT EVALUATION

PROPOSED SPECIFICATION

SECTION 3.1/4.1 REACTOR PROTECTION SYSTEM

Commonwealth Edison has evaluated the proposed amendment in accordance with the requirements of 10 CFR 51.21 and has determined that the amendment meets the requirements for categorical exclusion as specified by 10 CFR 51.22(c)(9). Commonwealth Edison has determined that the amendment involves no significant hazards consideration, there are no significant change in the types or significant increase in the amounts of any effluent that may be released offsite, and there is no significant increase in individual or cumulative occupational radiation exposure.

The proposed amendment does not modify the existing facility and does not involve any new operation of the plant. As such, the proposed amendment does not involve any change in the type or significant increases in effluents, or increase individual or cumulative occupational radiation exposure. The proposed amendment to Section 3.1/4.1, "Reactor Protection System" contains administrative changes such as including appropriate applicability statements within the specifications to define the applicability during operating mode and the required actions to be implemented in the event that specification cannot be met. The added requirements are based on Standard Technical Specifications and later operating plant requirements. The proposed specification also arranges the tables to provide for user-friendly presentation.

QC-1 / QC-2 DIFFERENCES

TS 3.1/4.1

"REACTOR PROTECTION SYSTEM"

COMPARISON OF UNIT 1 AND UNIT 2 TECHNICAL SPECIFICATIONS FOR THE IDENTIFICATION OF TECHNICAL DIFFERENCES

SECTION 3.1/4.1

REACTOR PROTECTION SYSTEM

Commonwealth Edison has conducted a comparison review of the Unit 1 and Unit 2 Technical Specifications to identify any technical differences in support of combining the Technical Specifications into one document. The intent of the review was not to identify any differences in presentacion style (e.g. table formats, use of capital letters, etc.), punctuation, or spelling errors but rather to identify areas which the Technical Specifications are technically or administratively different.

The review of Section 3.1/4.1 "Reactor Protection System" revealed the following technical differences:

Note [8] of "Notes for Tables 3.1-1, 3.1-2 and 3.1-3" (Page 3.1/4.1-14 for DPR-29) contains the statement, "1 inch on the water level instrumentation is > 504" above vessel zero (See Reference Bases 3.2)." which is not contained in the Unit 2 Technical Specification. This information is accurate for application on both units. Unit 1 and Unit 2 Technical Specification Bases section 3.2 contains the background for this statement.

Several administrative differences were identified as follows:

<u>Page 3.1/4.1-1</u>

Applicability	Unit 1:	Applies to instrumentation and ...
	Unit 2:	Applies to **the** instrumentation and...

<u>Page 3.1/4.1-3</u>

Paragraph 1	Unit 1:	c. minimize the energy which must be **absorbed** ...
	Unit 2:	c. minimize the energy which must be **adsorbed** ...

<u>Page 3.1/4.1-4</u>

Paragraph 2	Unit 1:	into **the** SDV. As indicated above, there...
	Unit 2:	into SDV. As indicated above, there...
Paragraph 3	Unit 1:	Loss of **condenser** vacuum...
	Unit 2:	Loss of **condensate** vacuum...

Page 3.1/4.1-5

Paragraph 7 Unit 1: discharge volume high level scrams are required for the **Start/Hot** Standby...
Unit 2: discharge volume high level scrams are required for the **Startup/Hot** Standby...

Page 3.1/4.1-7

Paragraph 2 Unit 1: The channels listed in **Table** ....
Unit 2: The channels listed in **Tables** ...

Page 3.1/4.1-8

Paragraph 2 Unit 1: rare occurrence and is detectable by an operator who observes that **one** signal...
Unit 2: rare occurrence and is detectable by an operator who observes than **on** signal....

Paragraph 4 Unit 1: amplifiers) are **predicted** to have ...
Unit 2: amplifiers) are **predicated** to have..

Paragraph 4 Unit 1: failures/hour. The bistable trip circuits are **predicted** to have ...
Unit 2: failures/hour. The bistable trip circuits are **predicated** to have ...

Paragraph 4 Unit 1: rate of less than 2 x 10 (-6) **failures/hours** ...
Unit 2: rate of less than 2 X 10 (-6) **failures/hour** ...

Paragraph 4 Unit 1: bistable trip circuits, the design reliability goal of 0.99999 is attained **with** ...
Unit 2: bistable trip circuits, the design reliability goal of 0.99999 is attained **wity** ...

Paragraph 3.1/4.1-9

Paragraph 2 Unit 1: once per month and, in addition, cross calibration checks of **the** flow ...
Unit 2: once per month and, in addition, cross calibration check of flow ...

Paragraph 3 Unit 1: subjected to a calibration/functional test of one month ...
Unit 2: subjected to a calibration/functional test **frequency** of one month ...

Page 3.1/4.1-10

Paragraph 1	Unit 1:	Group 3 devices are active only during a given portion of the **operation** cycle ...
	Unit 2:	Group 3 devices are active only during a given portion of the **operational** cycle ...
Paragraph 3	Unit 1:	and substations **indicate** that
	Unit 2:	and substations **indicates** that
Paragraph 4	Unit 1:	and approximately constant rate. Changes in **a** power distribution ...
	Unit 2:	and approximately constant rate. Changes in power distribution ...
Paragraph 4	Unit 1:	the APRM system every 7 days using heat balance data by calibrating ...
	Unit 2:	the APRM system every 7 days using heat balance data **and** by calibrating

Page 3.1/4.1-15

Note [9]	Unit 1:	electronic calibrator during the three month test ...
	Unit 2:	electronic calibrator during the three month **interval** test ...