

ATTACHMENT 3

**PROPOSED TECHNICAL  
SPECIFICATIONS**

Technical Specification 3/4.1

"REACTOR PROTECTION SYSTEM"

3.1 - LIMITING CONDITIONS FOR OPERATIONA. Reactor Protection System (RPS)

The reactor protection system (RPS) instrumentation CHANNEL(s) shown in Table 3.1.A-1 shall be OPERABLE.

APPLICABILITY:

As shown in Table 3.1.A-1.

ACTION:

1. With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per TRIP SYSTEM requirement for one TRIP SYSTEM, place the inoperable CHANNEL(s) and/or that TRIP SYSTEM in the tripped condition<sup>(a)(b)</sup> within 1 hour.
2. With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per TRIP SYSTEM requirement for both TRIP SYSTEM(s), place at least one TRIP SYSTEM in the tripped condition<sup>(b)(c)</sup> within 1 hour and take the ACTION required by Table 3.1.A-1.

4.1 - SURVEILLANCE REQUIREMENTSA. Reactor Protection System

1. Each reactor protection system instrumentation CHANNEL shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL MODE(s) and at the frequencies shown in Table 4.1.A-1.
2. LOGIC SYSTEM FUNCTIONAL TEST(s) and simulated automatic operation of all CHANNEL(s) shall be performed at least once per 18 months.
3. The response time of each reactor trip functional unit shown in Table 3.1.A-1 shall be demonstrated at least once per 18 months. Each test shall include at least one CHANNEL per TRIP SYSTEM such that all CHANNEL(s) are tested at least once every N times 18 months where N is the total number of redundant CHANNEL(s) in a specific reactor TRIP SYSTEM. The system response time for each trip function from the opening of the sensor contact up to and including the opening of the trip actuator shall not exceed 50 milliseconds.

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- a An inoperable CHANNEL need not be placed in the tripped condition when this would cause the trip function to occur. In these cases, the inoperable CHANNEL shall be restored to OPERABLE status within 2 hours or the ACTION required by Table 3.1.A-1 for that trip function shall be taken.
- b An inoperable CHANNEL or TRIP SYSTEM may be temporarily taken out of the tripped condition under administrative control for the purposes of conducting testing to: (1) determine the cause of the inoperability, or (2) to demonstrate OPERABILITY.
- c The TRIP SYSTEM need not be placed in the tripped condition if this would cause the trip function to occur. When a TRIP SYSTEM can be placed in the tripped condition without causing the trip function to occur, place the TRIP SYSTEM with the most inoperable CHANNEL(s) in the tripped condition; if both systems have the same number of inoperable CHANNEL(s), place either TRIP SYSTEM in the tripped condition.

TABLE 3.1.A-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION

<u>Functional Unit</u>	<u>Applicable OPERATIONAL MODE(s)</u>	<u>Minimum OPERABLE CHANNEL(s) per TRIP SYSTEM<sup>(a)</sup></u>	<u>ACTION</u>
1. Intermediate Range Monitor:			
a. Neutron Flux - High	2	3	11
	3, 4	2	12
	5 <sup>(c)</sup>	3	13
b. Inoperative	2	3	11
	3, 4	2	12
	5	3	13
2. Average Power Range Monitor <sup>(e)</sup> :			
a. Setdown Neutron Flux - High	2	2	11
	3	2	12
	5 <sup>(c,g)</sup>	2	13
b. Flow Biased Neutron Flux - High	1	2	14
c. Fixed Neutron Flux - High	1	2	14
d. Inoperative	1, 2	2	11
	3	2	12
	5 <sup>(g)</sup>	2	13
3. Reactor Vessel Steam Dome Pressure - High	1, 2 <sup>(f)</sup>	2	11
4. Reactor Vessel Water Level - Low	1, 2	2	11

TABLE 3.1.A-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

<u>Functional Unit</u>	<u>Applicable OPERATIONAL MODE(s)</u>	<u>Minimum OPERABLE CHANNEL(s) per TRIP SYSTEM<sup>(a)</sup></u>	<u>ACTION</u>
5. Main Steam Line Isolation Valve - Closure	1, 2 <sup>(j)</sup>	4	10
6. Main Steam Line Radiation - High	1, 2 <sup>(f)</sup>	2	15
7. Drywell Pressure - High	1, 2 <sup>(h)</sup>	2	11
8. Scram Discharge Volume Water Level - High			
a. $\Delta P$ Switch, and	1, 2	2	11
	5 <sup>(b,i)</sup>	2	13
b. Thermal Switch (Unit 2), or Float Switch (Unit 3)	1, 2	2	11
	5 <sup>(b,i)</sup>	2	13
9. Turbine Stop Valve - Closure	1 <sup>(d)</sup>	4	16
10. Turbine EHC Control Oil Pressure - Low	1 <sup>(d)</sup>	-	16
11. Turbine Control Valve Fast Closure	1 <sup>(d)</sup>	2	16
12. Turbine Condenser Vacuum - Low	1, 2 <sup>(j)</sup>	2	10



TABLE 3.1.A-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

<u>Functional Unit</u>	<u>Applicable OPERATIONAL MODE(s)</u>	<u>Minimum OPERABLE CHANNEL(s) per TRIP SYSTEM<sup>(a)</sup></u>	<u>ACTION</u>
13. Reactor Mode Switch Shutdown Position	1, 2	1	11
	3, 4	1	17
	5	1	13
14. Manual Scram	1, 2	1	11
	3, 4	1	18
	5	1	19

TABLE 3.1.A-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATIONACTION

- ACTION 10 - Be in at least STARTUP with reactor pressure less than 600 psig within 8 hours.
- ACTION 11 - Be in at least HOT SHUTDOWN within 12 hours.
- ACTION 12 - Verify all insertable control rods to be fully inserted in the core and lock the reactor mode switch in the Shutdown position within one hour.
- ACTION 13 - Suspend all operations involving CORE ALTERATIONS, and fully insert all insertable control rods within one hour. If SRM instrumentation is not OPERABLE per Specification 3.10.B, also suspend replacement of LPRMs.
- ACTION 14 - Be in at least STARTUP within 8 hours.
- ACTION 15 - Be in STARTUP with the main steam line isolation valves closed within 8 hours or in at least HOT SHUTDOWN within 12 hours.
- ACTION 16 - Initiate a reduction in THERMAL POWER within 15 minutes and reduce reactor power to less than 45% of RATED THERMAL POWER within 2 hours.
- ACTION 17 - Verify all insertable control rods to be fully inserted in the core within one hour.
- ACTION 18 - Lock the reactor mode switch in the Shutdown position within one hour.
- ACTION 19 - Suspend all operations involving CORE ALTERATIONS, and fully insert all insertable control rods and lock the reactor mode switch in the Shutdown position within one hour. If SRM instrumentation is not OPERABLE per Specification 3.10.B, also suspend replacement of LPRMs.

TABLE 3.1.A-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATIONTABLE NOTATION

- (a) A CHANNEL may be placed in an inoperable status for up to 2 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) This function may be bypassed, provided a control rod block is actuated, for reactor protection system logic reset in Refuel and Shutdown positions of the reactor mode switch.
- (c) Unless adequate SHUTDOWN MARGIN has been demonstrated per Specification 3/4.3.A and the "one-rod-out" Refuel mode switch interlock has been demonstrated OPERABLE per Specification 3.10.A, the "shorting links" shall be removed from the RPS circuitry prior to and during the time any control rod is withdrawn. However, this is not required for control rods removed per Specification 3.10.D or 3.10.E.
- (d) This function not required to be OPERABLE when THERMAL POWER is less than 45% of RATED THERMAL POWER.
- (e) An APRM CHANNEL is inoperable if there are fewer than 2 LPRM inputs per level or there are less than 50% of the normal complement of LPRM inputs to an APRM CHANNEL.
- (f) This function is not required to be OPERABLE when the reactor pressure vessel head is unbolted or removed per Specification 3.12.A.
- (g) Required to be OPERABLE only prior to and during required SHUTDOWN MARGIN demonstrations performed per Specification 3.12.B.
- (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) This function is not required to be OPERABLE when reactor pressure is less than 600 psig.

TABLE 4.1.A-1

## REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Functional Unit	Applicable OPERATIONAL MODE	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL <sup>(a)</sup> CALIBRATION
1. Intermediate Range Monitor:				
a. Neutron Flux - High	2 3, 4, 5	S, <sup>(b)</sup> S	S/U <sup>(c)</sup> , W <sup>(c)</sup> W	E <sup>(c)</sup> E
b. Inoperative	2, 3, 4, 5	NA	W <sup>(c)</sup>	NA
2. Average Power Range Monitor <sup>(f)</sup> :				
a. Setdown Neutron Flux - High	2 3, 5 <sup>(m)</sup>	S, <sup>(b)</sup> S	S/U <sup>(c)</sup> , W <sup>(c)</sup> W	SA <sup>(c)</sup> SA
b. Flow Biased Neutron Flux - High	1	S, D <sup>(g)</sup>	W	W <sup>(d,e)</sup> , SA
c. Fixed Neutron Flux - High	1	S	W	W <sup>(d)</sup> , SA
d. Inoperative	1, 2, 3, 5 <sup>(m)</sup>	NA	W	NA
3. Reactor Vessel Steam Dome Pressure - High	1, 2 <sup>(i)</sup>	NA	M	Q
4. Reactor Vessel Water Level - Low	1, 2	D	M	M <sup>(h)</sup>
5. Main Steam Line Isolation Valve - Closure	1, 2 <sup>(p)</sup>	NA	M	E
6. Main Steam Line Radiation - High	1, 2 <sup>(i)</sup>	S	M	E
7. Drywell Pressure - High	1, 2 <sup>(n)</sup>	NA	M	Q

TABLE 4.1.A-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

<u>Functional Unit</u>	<u>Applicable OPERATIONAL MODES</u>	<u>CHANNEL CHECK</u>	<u>CHANNEL FUNCTIONAL TEST</u>	<u>CHANNEL<sup>(a)</sup> CALIBRATION</u>
8. Scram Discharge Volume Water Level - High				
a. $\Delta P$ Switch, and	1, 2, 5 <sup>(j,k)</sup>	NA	Q	E
b. Thermal Switch (Unit 2), or Float Switch (Unit 3)	1, 2, 5 <sup>(j,k)</sup>	NA	Q	E
9. Turbine Stop Valve - Closure	1 <sup>(l)</sup>	NA	M	E
10. Turbine EHC Control Oil Pressure - Low	1 <sup>(l)</sup>	NA	M	Q
11. Turbine Control Valve Fast Closure	1 <sup>(l)</sup>	NA	M	E
12. Turbine Condenser Vacuum - Low	1, 2 <sup>(p)</sup>	NA	M	Q
13. Reactor Mode Switch Shutdown Position	1, 2, 3, 4, 5	NA	E	NA
14. Manual Scram	1, 2, 3, 4, 5	NA	M	NA

TABLE 4.1.A-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTSTABLE NOTATION

- (a) Neutron detectors may be excluded from the CHANNEL CALIBRATION.
- (b) The IRM and SRM channels shall be determined to overlap for at least ( $\frac{1}{2}$ ) decades during each startup after entering OPERATIONAL MODE 2 and the IRM and APRM channels shall be determined to overlap for at least ( $\frac{1}{2}$ ) decades during each controlled shutdown, if not performed within the previous 7 days.
- (c) Within 24 hours prior to startup, if not performed within the previous 7 days. The weekly CHANNEL FUNCTIONAL TEST may be used to fulfill this requirement.
- (d) This calibration shall consist of the adjustment of the APRM CHANNEL to conform, within 2% of RATED THERMAL POWER, to the power values calculated by a heat balance during OPERATIONAL MODE 1 when THERMAL POWER is  $\geq 25\%$  of RATED THERMAL POWER. This adjustment must be accomplished: a) within 2 hours if the APRM CHANNEL is indicating lower power values than the heat balance, or b) within 12 hours if the APRM CHANNEL is indicating higher power values than the heat balance. Until any required APRM adjustment has been accomplished, notification shall be posted on the reactor control panel.

Any APRM CHANNEL gain adjustment made in compliance with Specification 3.11.B shall not be included in determining the above difference. This calibration is not required when THERMAL POWER is  $< 25\%$  of RATED THERMAL POWER. The provisions of Specification 4.0.D are not applicable.

- (e) This calibration shall consist of the adjustment of the APRM flow biased channel to conform to a calibrated flow signal.
- (f) The LPRMs shall be calibrated at least once per 1000 effective full power hours (EFPH).
- (g) Verify measured recirculation loop flow to be greater than or equal to established recirculation loop flow at the existing pump speed.
- (h) Trip units are calibrated at least once per 31 days and transmitters are calibrated at least once per 18 months.
- (i) This function is not required to be OPERABLE when the reactor pressure vessel head is unbolted or removed per Specification 3.12.A.
- (j) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.10.D or 3.10.E.
- (k) This function may be bypassed, provided a control rod block is actuated, for reactor protection system reset in Refuel and Shutdown positions of the reactor mode switch.



TABLE 4.1.A-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

- (l) This function not required to be OPERABLE when THERMAL POWER is less than 45% of RATED THERMAL POWER.
- (m) Required to be OPERABLE only prior to and during required SHUTDOWN MARGIN demonstrations performed per Specification 3.12.B.
- (n) This function is not required to be OPERABLE when PRIMARY CONTAINMENT INTEGRITY is not required.
- (o) The provisions of Specification 4.0.D are not applicable to the CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION surveillances for entry into their OPERATIONAL MODE(s) from OPERATIONAL MODE 1 provided the surveillances are performed within 12 hours after such entry.
- (p) This function is not required to be OPERABLE when reactor pressure is less than 600 psig.



BASES3/4.1.A REACTOR PROTECTION SYSTEM INSTRUMENTATION

The reactor protection system (RPS) 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 perform its intended function, even during periods when instrument CHANNEL(s) may be out-of-service because of maintenance. When necessary, one CHANNEL may be made inoperable for brief intervals to conduct required surveillance.

The reactor protection system is made up of two independent TRIP SYSTEM(s), each having a minimum of two CHANNEL(s) of tripping devices. Each CHANNEL has an input from at least one instrument CHANNEL which monitors a critical parameter. The outputs of the CHANNEL(s) are combined in a one-out-of-two-logic, i.e., an input signal on either one or both of the CHANNEL(s) will cause a TRIP SYSTEM trip. The outputs of the TRIP SYSTEM(s) are arranged so that a trip on both systems is required to produce a reactor scram. This system meets the intent of the proposed 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). The bases for the trip settings of the RPS are discussed in the Bases for Specification 2.2.A.

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 result in the trip of the RPS CHANNEL. 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 non-coincident NMS reactor trip function logic is such that all channels go to both trip systems (1 out of n). However, 3 IRM CHANNELs and 2 APRM CHANNELs per TRIP SYSTEM are still required, i.e. a minimum of 6 IRMs and 4 APRMs.

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

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of the core and therefore are not required to have the capability to scram. If all control rods are 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. 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 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 (and is automatically bypassed) in OPERATIONAL MODE 1, the APRMs cover only the intermediate and power range; and the IRMs provide adequate coverage in the startup and intermediate range. The IRM inoperative function ensures that the instrument CHANNEL fails in the tripped condition upon loss of detector voltage.

Three APRM instrument CHANNEL(s) are provided for each TRIP SYSTEM. APRM CHANNEL(s) #1 and #3 operate contacts in one logic path and APRM CHANNEL(s) #2 and #3 operate contacts in the other logic path of the TRIP SYSTEM. APRM CHANNEL(s) #4, #5 and #6 are arranged similarly in the other TRIP SYSTEM's dual logic paths. Each 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 TRIP SYSTEM for maintenance, testing, or calibration. Additional IRM CHANNEL(s) have also been provided to allow for bypassing of one such CHANNEL.

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). A bypass in the Refuel or Startup/Hot Standby operational modes is provided for the turbine condenser low vacuum scram and main steam line isolation valve closure scrams 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.

The manual scram function is available in OPERATIONAL MODE(s) 1 through 5, thus providing for a manual means of rapidly inserting control rods whenever fuel is in the reactor.

The turbine stop valve closure scram, the turbine EHC control oil low pressure scram, and the turbine control valve fast closure scram occur by design on turbine first stage pressure which is normally equivalent to ~45% RATED THERMAL POWER. However, since this is dependent on bypass valve position, the conservative reactor power is used to determine applicability.

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 performance of the specified surveillances at the specified frequencies provides assurance that the protective functions associated with each CHANNEL can be completed as assumed in the safety analyses. A surveillance interval of "prior to startup" assures that these functions are available to perform their safety functions during control

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rod withdrawal, and hence these surveillances must be completed prior to initiating control rod withdrawal for the purpose of "approach to criticality".

3.1 - LIMITING CONDITIONS FOR OPERATION

A. Reactor Protection System (RPS)

The reactor protection system (RPS) instrumentation CHANNEL(s) shown in Table 3.1.A-1 shall be OPERABLE.

APPLICABILITY:

As shown in Table 3.1.A-1.

ACTION:

1. With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per TRIP SYSTEM requirement for one TRIP SYSTEM, place the inoperable CHANNEL(s) and/or that TRIP SYSTEM in the tripped condition<sup>(a)(b)</sup> within 1 hour.
2. With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per TRIP SYSTEM requirement for both TRIP SYSTEM(s), place at least one TRIP SYSTEM in the tripped condition<sup>(b)(c)</sup> within 1 hour and take the ACTION required by Table 3.1.A-1.

4.1 - SURVEILLANCE REQUIREMENTS

A. Reactor Protection System

1. Each reactor protection system instrumentation CHANNEL shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL MODE(s) and at the frequencies shown in Table 4.1.A-1.
2. LOGIC SYSTEM FUNCTIONAL TEST(s) and simulated automatic operation of all CHANNEL(s) shall be performed at least once per 18 months.
3. The response time of each reactor trip functional unit shown in Table 3.1.A-1 shall be demonstrated at least once per 18 months. Each test shall include at least one CHANNEL per TRIP SYSTEM such that all CHANNEL(s) are tested at least once every N times 18 months where N is the total number of redundant CHANNEL(s) in a specific reactor TRIP SYSTEM. The system response time for each trip function from the opening of the sensor contact up to and including the opening of the trip actuator shall not exceed 50 milliseconds.

- a. An inoperable CHANNEL need not be placed in the tripped condition when this would cause the Trip Function to occur. In these cases, the inoperable CHANNEL shall be restored to OPERABLE status within 2 hours or the ACTION required by Table 3.1.A-1 for that Trip Function shall be taken.
- b. An inoperable CHANNEL or TRIP SYSTEM may be temporarily taken out of the tripped condition under administrative control for the purposes of conducting testing to: (1) determine the cause of the inoperability, or (2) to demonstrate OPERABILITY.
- c. The TRIP SYSTEM need not be placed in the tripped condition if this would cause the Trip Function to occur. When a TRIP SYSTEM can be placed in the tripped condition without causing the Trip Function to occur, place the TRIP SYSTEM with the most inoperable CHANNEL(s) in the tripped condition; if both systems have the same number of inoperable CHANNEL(s), place either TRIP SYSTEM in the tripped condition.



TABLE 3.1.A-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION

<u>Functional Unit</u>	<u>Applicable OPERATIONAL MODE(s)</u>	<u>Minimum OPERABLE CHANNEL(s) per TRIP SYSTEM<sup>(a)</sup></u>	<u>ACTION</u>
1. Intermediate Range Monitor:			
a. Neutron Flux - High	2	3	11
	3, 4	2	12
	5 <sup>(c)</sup>	3	13
b. Inoperative	2	3	11
	3, 4	2	12
	5	3	13
2. Average Power Range Monitor <sup>(a)</sup> :			
a. Setdown Neutron Flux - High	2	2	11
	3	2	12
	5 <sup>(c,g)</sup>	2	13
b. Flow Biased Neutron Flux - High	1	2	14
c. Fixed Neutron Flux - High	1	2	14
d. Inoperative	1, 2	2	11
	3	2	12
	5 <sup>(g)</sup>	2	13
3. Reactor Vessel Steam Dome Pressure - High	1, 2 <sup>(f)</sup>	2	11
4. Reactor Vessel Water Level - Low	1, 2	2	11

TABLE 3.1.A-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

<u>Functional Unit</u>	<u>Applicable OPERATIONAL MODE(s)</u>	<u>Minimum OPERABLE CHANNEL(s) per TRIP SYSTEM<sup>(a)</sup></u>	<u>ACTION</u>
5. Main Steam Line Isolation Valve - Closure	1	4	14
6. Main Steam Line Radiation - High	1, 2 <sup>(f)</sup>	2	15
7. Drywell Pressure - High	1, 2 <sup>(h)</sup>	2	11
8. Scram Discharge Volume Water Level - High			
a. $\Delta P$ Switch, and	1, 2	2	11
	5 <sup>(b,i)</sup>	2	13
b. Thermal Switch	1, 2	2	11
	5 <sup>(b,i)</sup>	2	13
9. Turbine Stop Valve - Closure	1 <sup>(d)</sup>	4	16
10. Turbine EHC Control Oil Pressure - Low	1 <sup>(d)</sup>	2	16
11. Turbine Control Valve Fast Closure	1 <sup>(d)</sup>	2	16
12. Turbine Condenser Vacuum - Low	1	2	14

TABLE 3.1.A-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

<u>Functional Unit</u>	<u>Applicable OPERATIONAL MODE(s)</u>	<u>Minimum OPERABLE CHANNEL(s) per TRIP SYSTEM<sup>(a)</sup></u>	<u>ACTION</u>
13. Reactor Mode Switch Shutdown Position	1, 2	1	11
	3, 4	1	17
	5	1	13
14. Manual Scram	1, 2	1	11
	3, 4	1	18
	5	1	19



TABLE 3.1.A-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATIONACTION

- ACTION 11 - Be in at least HOT SHUTDOWN within 12 hours.
- ACTION 12 - Verify all insertable control rods to be fully inserted in the core and lock the reactor mode switch in the Shutdown position within one hour.
- ACTION 13 - Suspend all operations involving CORE ALTFRACTIONS, and fully insert all insertable control rods within one hour. If SRM instrumentation is not OPERABLE per Specification 3.10.B, also suspend replacement of LPRMs.
- ACTION 14 - Be in at least STARTUP within 8 hours.
- ACTION 15 - Be in STARTUP with the main steam line isolation valves closed within 8 hours or in at least HOT SHUTDOWN within 12 hours.
- ACTION 16 - Initiate a reduction in THERMAL POWER within 15 minutes and reduce reactor power to less than 45% of RATED THERMAL POWER within 2 hours.
- ACTION 17 - Verify all insertable control rods to be fully inserted in the core within one hour.
- ACTION 18 - Lock the reactor mode switch in the Shutdown position within one hour.
- ACTION 19 - Suspend all operations involving CORE ALTERATIONS, and fully insert all insertable control rods and lock the reactor mode switch in the Shutdown position within one hour. If SRM instrumentation is not OPERABLE per Specification 3.10.B, also suspend replacement of LPRMs.

TABLE 3.1.A-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATIONTABLE NOTATION

- (a) A CHANNEL may be placed in an inoperable status for up to 2 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) This function may be bypassed, provided a control rod block is actuated, for reactor protection system logic reset in Refuel and Shutdown positions of the reactor mode switch.
- (c) Unless adequate SHUTDOWN MARGIN has been demonstrated per Specification 3/4.3.A and the "one-rod-out" Refuel mode switch interlock has been demonstrated OPERABLE per Specification 3.10.A, the "shorting links" shall be removed from the RPS circuitry prior to and during the time any control rod is withdrawn. However, this is not required for control rods removed per Specification 3.10.D or 3.10.E.
- (d) This function not required to be OPERABLE when THERMAL POWER is less than 45% of RATED THERMAL POWER.
- (e) An APRM CHANNEL is inoperable if there are fewer than 2 LPRM inputs per level or there are less than 50% of the normal complement of LPRM inputs to an APRM CHANNEL.
- (f) This function is not required to be OPERABLE when the reactor pressure vessel head is unbolted or removed per Specification 3.12.A.
- (g) Required to be OPERABLE only prior to and during required SHUTDOWN MARGIN demonstrations performed per Specification 3.12.B.
- (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.

TABLE 4.1.A-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

<u>Functional Unit</u>	<u>Applicable OPERATIONAL MODES</u>	<u>CHANNEL CHECK</u>	<u>CHANNEL FUNCTIONAL TEST</u>	<u>CHANNEL<sup>(a)</sup> CALIBRATION</u>
1. Intermediate Range Monitor:				
a. Neutron Flux - High	2 3, 4, 5	S, <sup>(b)</sup> S	S/U <sup>(c)</sup> , W <sup>(o)</sup> W	E <sup>(o)</sup> E
b. Inoperative	2, 3, 4, 5	NA	W <sup>(o)</sup>	NA
2. Average Power Range Monitor <sup>(f)</sup> :				
a. Setdown Neutron Flux - High	2 3, 5 <sup>(m)</sup>	S, <sup>(b)</sup> S	S/U <sup>(c)</sup> , W <sup>(o)</sup> W	SA <sup>(o)</sup> SA
b. Flow Biased Neutron Flux - High	1	S, D <sup>(g)</sup>	W	W <sup>(d,e)</sup> , SA
c. Fixed Neutron Flux - High		S	W	W <sup>(d)</sup> , SA
d. Inoperative	1, 2, 3, 5 <sup>(m)</sup>	NA	W	NA
3. Reactor Vessel Steam Dome Pressure - High	1, 2 <sup>(i)</sup>	NA	M	Q
4. Reactor Vessel Water Level - Low	1, 2	D	M	M <sup>(h)</sup>
5. Main Steam Line Isolation Valve - Closure	1	NA	M	E
6. Main Steam Line Radiation - High	1, 2 <sup>(i)</sup>	S	M	E
7. Drywell Pressure - High	1, 2 <sup>(n)</sup>	NA	M	Q

TABLE 4.1.A-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

<u>Functional Unit</u>	<u>Applicable OPERATIONAL MODES</u>	<u>CHANNEL CHECK</u>	<u>CHANNEL FUNCTIONAL TEST</u>	<u>CHANNEL<sup>(a)</sup> CALIBRATION</u>
8. Scram Discharge Volume Water Level - High				
a. $\Delta P$ Switch, and	1, 2, 5 <sup>(j,k)</sup>	NA	Q	E
b. Thermal Switch	1, 2, 5 <sup>(j,k)</sup>	NA	Q	E
9. Turbine Stop Valve - Closure	1 <sup>(l)</sup>	NA	M	E
10. Turbine EHC Control Oil Pressure - Low	1 <sup>(l)</sup>	NA	M	Q
11. Turbine Control Valve Fast Closure	1 <sup>(l)</sup>	NA	M	E
12. Turbine Condenser Vacuum - Low	1	NA	M	Q
13. Reactor Mode Switch Shutdown Position	1, 2, 3, 4, 5	NA	E	NA
14. Manual Scram	1, 2, 3, 4, 5	NA	M	NA

TABLE 4.1.A-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTSTABLE NOTATION

- (a) Neutron detectors may be excluded from the CHANNEL CALIBRATION.
- (b) The IRM and SRM channels shall be determined to overlap for at least ( $\frac{1}{2}$ ) decades during each startup after entering OPERATIONAL MODE 2 and the IRM and APRM channels shall be determined to overlap for at least ( $\frac{1}{4}$ ) decades during each controlled shutdown, if not performed within the previous 7 days.
- (c) Within 24 hours prior to startup, if not performed within the previous 7 days. The weekly CHANNEL FUNCTIONAL TEST may be used to fulfill this requirement.
- (d) This calibration shall consist of the adjustment of the APRM CHANNEL to conform, within 2% of RATED THERMAL POWER, to the power values calculated by a heat balance during OPERATIONAL MODE 1 when THERMAL POWER is  $\geq 25\%$  of RATED THERMAL POWER. This adjustment must be accomplished: a) within 2 hours if the APRM CHANNEL is indicating lower power values than the heat balance, or b) within 12 hours if the APRM CHANNEL is indicating higher power values than the heat balance. Until any required APRM adjustment has been accomplished, notification shall be posted on the reactor control panel.

Any APRM CHANNEL gain adjustment made in compliance with Specification 3.11.B shall not be included in determining the above difference. This calibration is not required when THERMAL POWER is  $< 25\%$  of RATED THERMAL POWER. The provisions of Specification 4.0.D are not applicable.

- e) This calibration shall consist of the adjustment of the APRM flow biased channel to conform to a calibrated flow signal.
- (f) The LPRMs shall be calibrated at least once per 1000 effective full power hours (EFPH).
- (g) Verify measured recirculation loop flow to be greater than or equal to established recirculation loop flow at the existing pump speed.
- (h) Trip units are calibrated at least once per 31 days and transmitters are calibrated at least once per 18 months.
- (i) This function is not required to be OPERABLE when the reactor pressure vessel head is unbolted or removed per Specification 3.12.A.
- (j) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.10.D or 3.10.E.
- (k) This function may be bypassed, provided a control rod block is actuated, for reactor protection system reset in Refuel and Shutdown positions of the reactor mode switch.

TABLE 4.1.A-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

- (l) This function not required to be OPERABLE when THERMAL POWER is less than 45% of RATED THERMAL POWER.
- (m) Required to be OPERABLE only prior to and during required SHUTDOWN MARGIN demonstrations performed per Specification 3.12.B.
- (n) This function is not required to be OPERABLE when PRIMARY CONTAINMENT INTEGRITY is not required.
- (o) The provisions of Specification 4.0.D are not applicable to the CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION surveillances for entry into their OPERATIONAL MODE(s) from OPERATIONAL MODE 1 provided the surveillances are performed within 12 hours after such entry.



BASES3/4.1.A REACTOR PROTECTION SYSTEM INSTRUMENTATION

The reactor protection system (RPS) 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 perform its intended function, even during periods when instrument CHANNEL(s) may be out-of-service because of maintenance. When necessary, one CHANNEL may be made operable for brief intervals to conduct required surveillance.

The reactor protection system is made up of two independent TRIP SYSTEM(s), each having a minimum of two CHANNEL(s) of tripping devices. Each CHANNEL has an input from at least one instrument CHANNEL which monitors a critical parameter. The outputs of the CHANNEL(s) are combined in a one-out-of-two-logic, i.e., an input signal on either one or both of the CHANNEL(s) will cause a TRIP SYSTEM trip. The outputs of the TRIP SYSTEM(s) are arranged so that a trip on both systems is required to produce a reactor scram. This system meets the intent of the proposed 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). The bases for the trip settings of the RPS are discussed in the Bases for Specification 2.2.A.

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 result in the trip of the RPS CHANNEL. 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 non-coincident NMS reactor trip function logic is such that all channels go to both trip systems (1 out of n). However, 3 IRM CHANNELS and 2 APRM CHANNELS per TRIP SYSTEM are still required, i.e. a minimum of 6 IRMs and 4 APRMs.

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



## BASES

of the core and therefore are not required to have the capability to scram. If all control rods are 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. 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 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 (and is automatically bypassed) in OPERATIONAL MODE 1, the APRMs cover only the intermediate and power range; and the IRMs provide adequate coverage in the startup and intermediate range. The IRM inoperative function ensures that the instrument CHANNEL fails in the tripped condition upon loss of detector voltage.

Three APRM instrument CHANNEL(s) are provided for each TRIP SYSTEM. APRM CHANNEL(s) #1 and #3 operate contacts in one logic path and APRM CHANNEL(s) #2 and #3 operate contacts in the other logic path of the TRIP SYSTEM. APRM CHANNEL(s) #4, #5 and #6 are arranged similarly in the other TRIP SYSTEM's dual logic paths. Each 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 TRIP SYSTEM for maintenance, testing, or calibration. Additional IRM CHANNEL(s) have also been provided to allow for bypassing of one such CHANNEL.

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). A bypass in the Refuel or Startup/Hot Standby operational modes is provided for the turbine condenser low vacuum scram and main steam line isolation valve closure scrams 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.

The manual scram function is available in OPERATIONAL MODE(s) 1 through 5, thus providing for a manual means of rapidly inserting control rods whenever fuel is in the reactor.

The turbine stop valve closure scram, the turbine EHC control oil low pressure scram, and the turbine control valve fast closure scram occur by design on turbine first stage pressure which is normally equivalent to ~45% RATED THERMAL POWER. However, since this is dependent on bypass valve position, the conservative reactor power is used to determine applicability.

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 performance of the specified surveillances at the specified frequencies provides assurance that the protective functions associated with each CHANNEL can be completed as assumed in the safety analyses. A surveillance interval of "prior to startup" assures that these functions are available to perform their safety functions during control

BASES

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rod withdrawal, and hence these surveillances must be completed prior to initiating control rod withdrawal for the purpose of "approach to criticality".

ATTACHMENT 4

**EXISTING TECHNICAL  
SPECIFICATIONS**

Technical Specification 3/4.1

"REACTOR PROTECTION SYSTEM"

## ATTACHMENT 4

### DELETION OF CURRENT TECHNICAL SPECIFICATIONS

This technical specification amendment will replace the current section 3.1/4.1, Reactor Protection System, for the Dresden Unit 2 and Unit 3 Technical Specifications. The specifications are replaced in its entirety with revised pages that combine the Unit 2 and Unit 3 specifications.

Delete the following pages:

DPR - 19	DPR - 25
3/4.1-1	3/4.1-1
3/4.1-2	3/4.1-2
3/4.1-3	3/4.1-3
3/4.1-4	3/4.1-4
3/4.1-5	3/4.1-5
3/4.1-6	3/4.1-6
3/4.1-7	3/4.1-7
3/4.1-8	3/4.1-8
3/4.1-9	3/4.1-9
3/4.1-10	3/4.1-10
B 3/4.1-11	B 3/4.1-11
B 3/4.1-12	B 3/4.1-12
B 3/4.1-13	B 3/4.1-13
B 3/4.1-14	B 3/4.1-14
B 3/4.1-15	B 3/4.1-15
B 3/4.1-16	B 3/4.1-16
B 3/4.1-17	B 3/4.1-17
B 3/4.1-18	B 3/4.1-18
B 3/4.1-19	B 3/4.1-19
B 3/4.1-20	B 3/4.1-20

# ATTACHMENT 4

## DELETION OF CURRENT TECHNICAL SPECIFICATIONS

This technical specification amendment will replace the current section 3.1/4.1, Reactor Protection System, for the Quad Cities Unit 1 and Unit 2 Technical Specifications. The specifications are replaced in its entirety with revised pages that combine the Unit 1 and Unit 2 specifications.

Delete the following pages:

DPR - 29	DPR - 30
3.1/4.1-1	3.1/4.1-1
3.1/4.1-2	3.1/4.1-2
3.1/4.1-3	3.1/4.1-2a
3.1/4.1-4	3.1/4.1-3
3.1/4.1-5	3.1/4.1-4
3.1/4.1-6	3.1/4.1-5
3.1/4.1-7	3.1/4.1-6
3.1/4.1-8	3.1/4.1-7
3.1/4.1-9	3.1/4.1-7a
3.1/4.1-10	3.1/4.1-7b
3.1/4.1-11	3.1/4.1-8
3.1/4.1-12	3.1/4.1-9
3.1/4.1-13	3.1/4.1-10
3.1/4.1-14	3.1/4.1-11
3.1/4.1-15	3.1/4.1-11a
3.1/4.1-16	3.1/4.1-12
3.1/4.1-17	3.1/4.1-13
Figure 4.1-1	3.1/4.1-14
	3.1/4.1-15
	Figure 4.1-1



ATTACHMENT 5

**DRESDEN 2/3 DIFFERENCES**

Technical Specification 3/4.1

"REACTOR PROTECTION SYSTEM"

## ATTACHMENT 5

### COMPARISON OF DRESDEN UNIT 2 AND UNIT 3 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 Dresden Unit 2 and Unit 3 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 presentation 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:

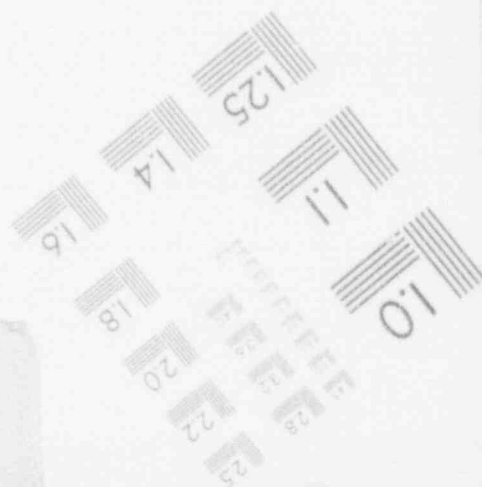
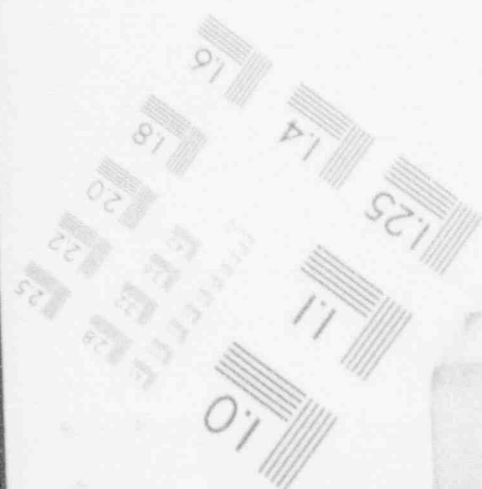
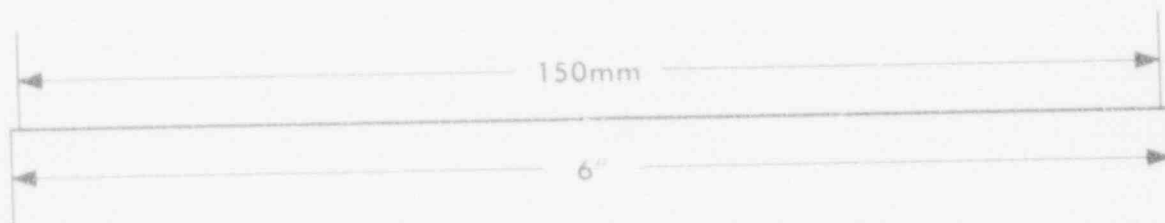
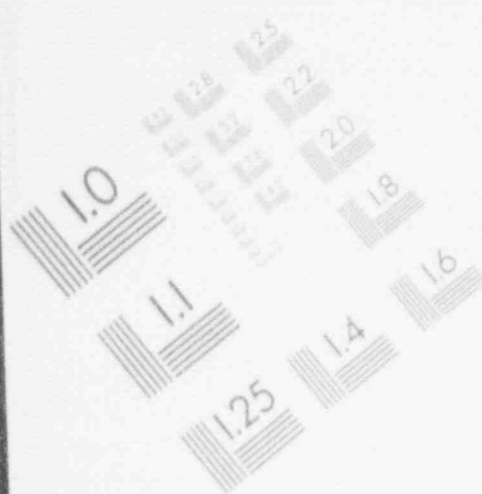
Unit 2 uses thermal switches to determine high water level in the Scram Discharge Volume while Unit 3 uses float switches. This affects Table 3.1.1, Table 4.1.1, Note 7 to Table 4.1.1, and the bases for the Scram Discharge Volume section.

Unit 2 has a Hydrogen Addition system. This affects Table 3.1.1 Note 11 and the bases for the Main Steam Line Rad Monitor setpoint.



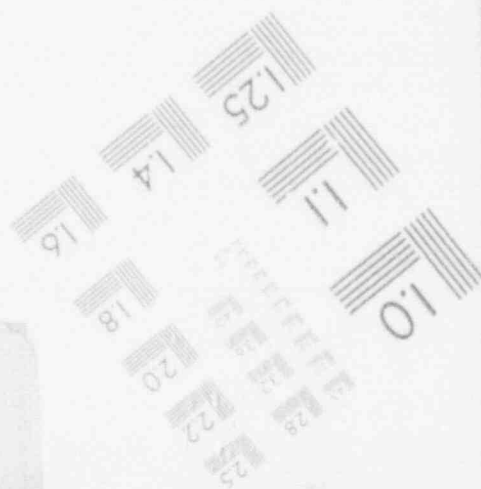
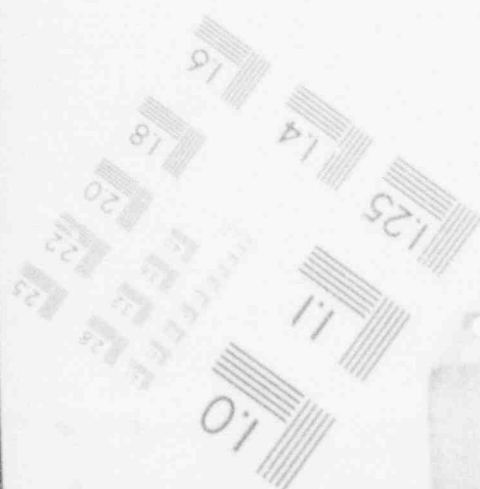
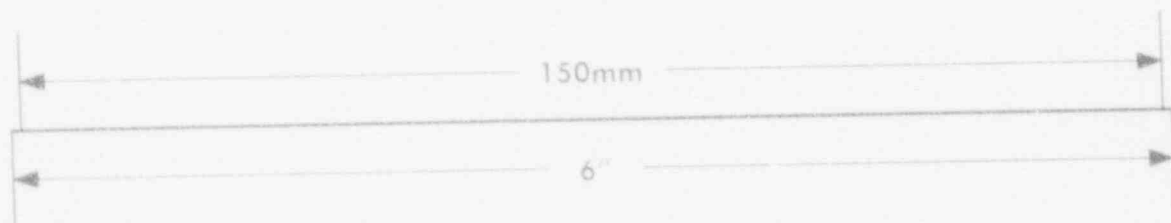
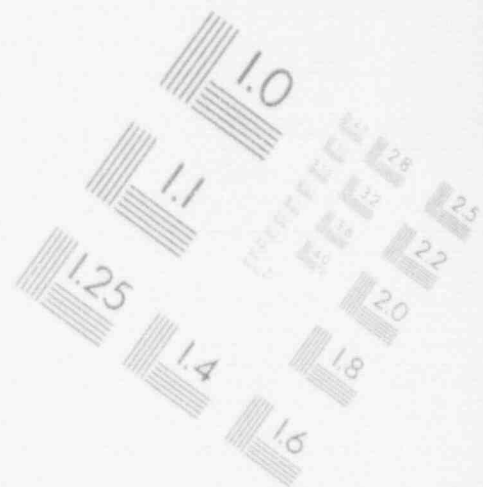
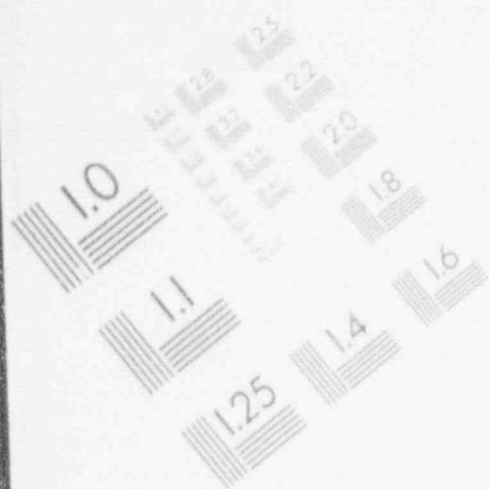
1

IMAGE EVALUATION  
TEST TARGET (MT-3)



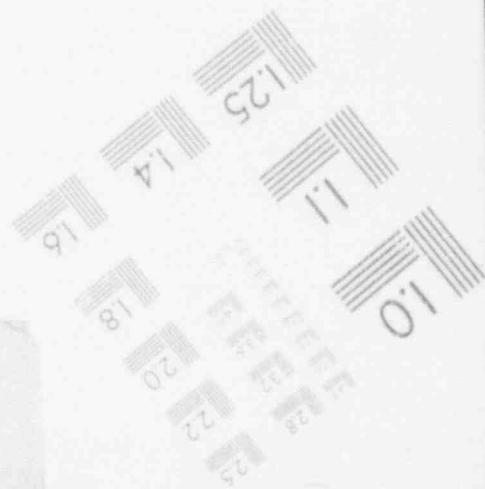
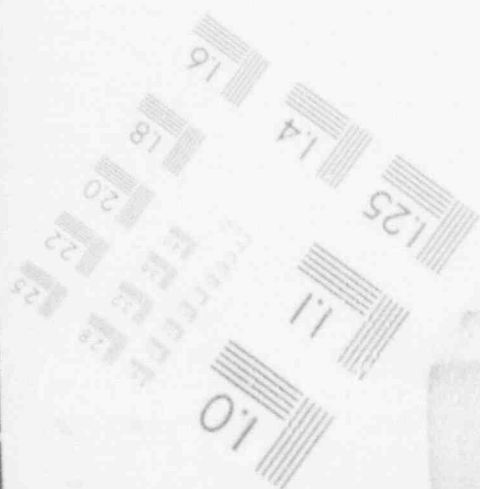
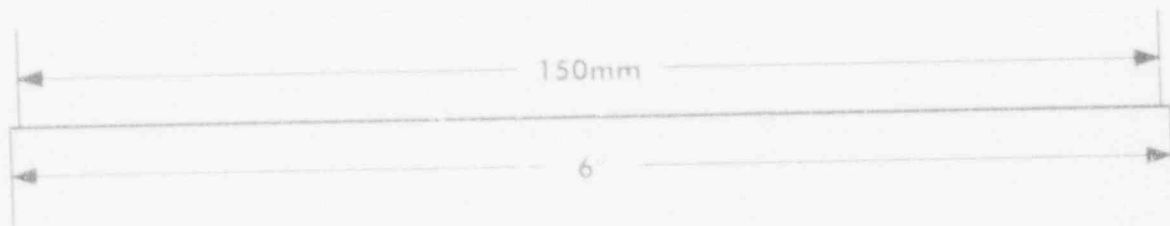
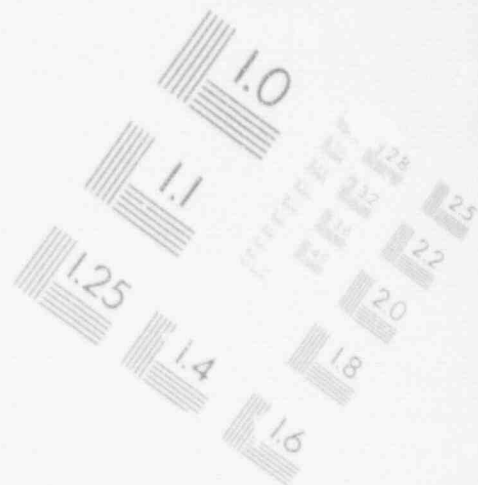
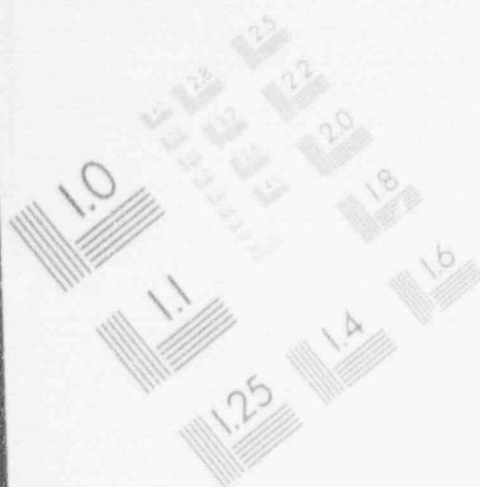
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IMAGE EVALUATION  
TEST TARGET (MT-3)



1

IMAGE EVALUATION  
TEST TARGET (MT-3)



ATTACHMENT 5

**QUAD CITIES 1/2 DIFFERENCES**

Technical Specification 3/4.1

"REACTOR PROTECTION SYSTEM"

## ATTACHMENT 5

### COMPARISON OF QUAD CITIES 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 Quad Cities 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 presentation 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.

ATTACHMENT 6

**SIGNIFICANT HAZARDS  
CONSIDERATIONS AND  
ENVIRONMENTAL ASSESSMENT  
EVALUATION**

Technical Specification 3/4.1

"REACTOR PROTECTION SYSTEM"



## ATTACHMENT 6

### EVALUATION OF SIGNIFICANT HAZARDS CONSIDERATION

Commonwealth Edison has evaluated this proposed amendment and determined that it involves no significant hazards consideration. According to 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; or
- 2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or
- 3) Involve a significant reduction in a margin of safety.

The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated because:

In general, the proposed changes represent the conversion of current requirements to a more generic format, or the addition of requirements which are based on the current safety analysis. Implementation of these changes will provide increased reliability of equipment assumed to operate in the current safety analysis, or provide continued assurance that specified parameters remain within their acceptance limits, and as such, will not significantly increase the probability or consequences of a previously evaluated accident.

Some of the proposed changes represent minor curtailments of the current requirements which are based on generic guidance or previously approved provisions for other stations. These proposed changes are consistent with the current safety analyses and have been previously determined to represent sufficient requirements for the assurance of reliability of equipment assumed to operate in the safety analysis, or provide continued assurance that specified parameters remain within their acceptance limits. As such, these changes will not significantly increase the probability or consequences of a previously evaluated accident.

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.

Several changes were made from the existing specification requirements for equipment operability in the Refuel and Hot Standby modes. The changes were made in accordance with the Standard Technical Specifications and combine the requirements of the present specifications and notes to the specifications to have equipment available that will not provide any function in the applicable modes.

## ATTACHMENT 6

Action 14 to table 3.1.A-1 was changed from the STS to retain the present 8 hour time requirement to be in the Startup mode. The retention of the 8 hour time allotment allows for a controlled reactor shutdown.

A line item improvement adopted from Perry Nuclear Power Station allows for the movement from operational mode 1 to operational mode 2 prior to performing all of the required surveillances provided that they are performed within the subsequent 12 hours.

Not requiring the APRMs to be operable in operational mode 5 will not increase the probability of inadvertent reactor criticality during refueling operations. IRMs, SRMs, refueling interlocks and procedural restrictions provide assurance that inadvertent criticality does not occur. The consequences of an accident will not be increased by the proposed change because of existing equipment and procedural restrictions which prevent an inadvertent criticality event during refueling. Therefore, the proposed changes do not result in an increase in the probability or consequence of an accident previously evaluated.

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

In general, the proposed changes represent the conversion of current requirements to a more generic format, or the addition of requirements which are based on the current safety analysis. Others represent minor curtailments of the current requirements which are based on generic guidance or previously approved provisions for other stations. These changes do not involve revisions to the design of the station. Some of the changes may involve revision in the operation of the station; however, these changes provide additional restrictions which are in accordance with the current safety analyses, or are to provide for additional testing or surveillances which will not introduce new failure mechanisms beyond those already considered in the current safety analyses. Therefore, these changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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.

The current specifications allow 8 hours to be in the startup mode and thus, retaining the 8 hour requirement will not introduce a new or different kind of accident.

Allowing the movement from operational mode 1 to operational mode 2 prior to performing the all of the required surveillances

The proposed changes to the Technical Specification will remove the APRM operability requirement while in operational mode 5 (except during shutdown margin

## ATTACHMENT 6

demonstration testing); however, the SRMs and the IRMs will still be required to be operable in operational mode 5. The IRMs are designed to detect and respond to increases in neutron flux within the local core regions.

No new types of accidents would be introduced since the SRMs and IRMs are available and required to be operable in operational mode 5. Both SRMs and IRMs would indicate and provide a control rod block or scram signal, as appropriate, to an increase in neutron flux to mitigate a transient event. The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

Involve a significant reduction in the margin of safety because:

In general, the proposed changes represent the conversion of current requirements to a more generic format, or the addition of requirements which are based on the current safety analysis. Others represent minor curtailments of the current requirements which are based on generic guidance or previously approved provisions for other stations. Some of the latter individual items may introduce minor reductions in the margin of safety when compared to the current requirements. However, other individual changes are the adoption of new requirements which will provide significant enhancement of the reliability of the equipment assumed to operate in the safety analysis, or provide enhanced assurance that specified parameters remain within their acceptance limits. These enhancements compensate for the individual minor reductions, such that taken together, the proposed changes will not significantly reduce the margin of safety.

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.

The changes to the Tables in Section 3/4.1 follow proven STS guidelines that have been implemented at other operating BWR plants. These changes have been evaluated for use at Dresden and 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.

## ATTACHMENT 6

### ENVIRONMENTAL ASSESSMENT STATEMENT APPLICABILITY REVIEW

Commonwealth Edison has evaluated the proposed amendment against the criteria for the identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.20. It has been determined that the proposed changes meet the criteria for a categorical exclusion as provided under 10 CFR 51.22 (c)(9). This conclusion has been determined because the changes requested do not pose significant hazards consideration or do not involve a significant increase in the amounts, and no significant changes in the types, of any effluents that may be released offsite. Additionally, this request does not involve a significant increase in individual or cumulative occupational radiation exposure. Therefore, the Environmental Assessment Statement is not applicable for these changes.

ATTACHMENT 7

**GENERIC LETTER 87-09  
IMPLEMENTATION**

Technical Specification 3/4.1

"REACTOR PROTECTION SYSTEM"

## ATTACHMENT 7

### APPLICATION OF GENERIC LETTER 87-09 REVISION TO PROPOSED SPECIFICATION 3.0.D

The Dresden/Quad Cities Technical Specification Upgrade Program has implemented the recommendations of Generic Letter 87-09. Included in these recommendations was a revision to Standard Technical Specification 3.0.4 for which these stations had no corresponding restriction. Under the proposed Specification, entry into an operational mode or other specified condition is permitted under compliance with the Action requirements. Indicated below is the method of implementation for this recommendation for each Action requirement in this package.

PROPOSED TECH SPEC	ACTION	APPL MODEs	CONT. OPS IN APP. COND?	CAT	CLARIFICATION
3.1.1	1	VAR	UNLIMITED	OK	
	2	VAR	PER TABLE	--	Must check Table applicable ACTIONs
T3.1.A-1	10	1-2	8 hrs	NO	Must exit APP MODE
	11	1-2	12 hrs	NO	Must exit APP MODE
	12	3-4	UNLIMITED	OK	Do not have to exit Applicable Modes
	13	5	UNLIMITED	OK	Do not have to exit Applicable Mode
	14	1	8 hrs	NO	Must exit APP MODE
	15	1-2	UNLIMITED	OK	For MSL high rad monitor with MSIVs closed only
	15	1	6 hrs	NO	For low condenser vacuum
	16	1, >45%	2 hrs	NO	Must exit APP MODE
	17	3-4	UNLIMITED	OK	Do not have to exit Applicable Modes
	18	3-4	UNLIMITED	OK	Do not have to exit Applicable Modes
	19	5	UNLIMITED	OK	Do not have to exit Applicable Modes