ATTACHMENT 3

PROPOSED TECHNICAL SPECIFICATIONS

Technical Specification 1.0

"DEFINITIONS"

1.0 DEFINITIONS

The following terms are defined so that uniform interpretation of these specifications may be achieved. The defined terms appear in capitalized type and shall be applicable throughout these Technical Specifications.

ACTION

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

AVERAGE PLANAR EXPOSURE (APE)

The AVERAGE PLANAR EXPOSURE (APE) shall be applicable to a specific planar height and is equal to the sum of the exposure of all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle.

AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) shall be applicable to a specific planar height and is equal to the sum of the LINEAR HEAT GENERATION RATE(s) for all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle.

CHANNEL

A CHANNEL shall be an arrangement of a sensor and associated components used to evaluate plant variables and generate a single protective action signal. A CHANNEL terminates and loses its identity where single action signals are combined in a TRIP SYSTEM or logic system.

CHANNEL CALIBRATION

A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the CHANNEL output such that it responds with the necessary range and accuracy to known values of the parameter which the CHANNEL monitors. The CHANNEL CALIBRATION shall encompass the entire CHANNEL including the required sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping or total CHANNEL steps such that the entire CHANNEL is calibrated.

CHANNEL CHECK

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

1.0 DEFINITIONS

CHANNEL FUNCTIONAL TEST

A CHANNEL FUNCTIONAL TEST shall be:

a. Analog CHANNEL(s) - the injection of a simulated signal into the CHANNEL as close to the sensor as practicable to verify OPERABILITY including required alarm and/or trip functions and CHANNEL failure trips.

b. Bistable CHANNEL(s) - the injection of a simulated signal into the sensor to verify OPERABILITY including required alarm and/or trip functions.

The CHANNEL FUNCTIONAL TEST may be performed by any series of sequential, overlapping or total CHANNEL steps such that the entire CHANNEL is tested.

CORE ALTERATION

CORE ALTERATION shall be the addition, removal, relocation or movement of fuel, sources, incore instruments or reactivity controls within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Normal movement (including replacement) of the SRMs, IRMs, TIPs, LPRMs, or special movable detectors is not considered a CORE ALTERATION. Suspension of CORE ALTERATION(s) shall not preclude completion of the movement of a component to a safe conservative position.

CORE OPERATING LIMITS REPORT (COLR)

The CORE OPERATING LIMITS REPORT (COLR) shall be the unit specific document that provides core operating limits for the current operating cycle. These cycle specific core operating limits shall be determined for each operating cycle in accordance with Specification 6.6. Plant operation within these operating limits is addressed in individual specifications.

CRITICAL POWER RATIO (CPR)

The CRITICAL POWER RATIO (CPR) shall be the ratio of that power in the assembly which is calculated by application of the applicable NRC approved critical power correlation to cause some point in the assembly to experience transition boiling, divided by the actual assembly power.

DOSE EQUIVALENT I-131

DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcurie/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors For Power and Test Reactor Sites."

FRACTION OF RATED THERMAL POWER (FRTP)

The FRACTION OF RATED THERMAL POWER (FRTP) shall be the measured THERMAL POWER divided by the RATED THERMAL POWER.

1.0 DEFINITIONS

FREQUENCY NOTATION

The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1-1.

FUEL DESIGN LIMITING RATIO (FDLRX)

The FUEL DESIGN LIMITING RATIO (FDLRX) shall be the limit used to assure that the fuel operates within the end-of-life steady-state design criteria by, among other items, limiting the release of fission gas to the cladding plenum.

FUEL DESIGN LIMITING RATIO for CENTERLINE MELT (FDLRC)

The FUEL DESIGN LIMITING RATIO for CENTERLINE MELT (FDLRC) shall be the limit used to assure that the fuel will neither experience centerline melt nor exceed 1% plastic cladding strain for transient overpower events beginning at any power and terminating at 120% of RATED THERMAL POWER.

IDENTIFIED LEAKAGE

IDENTIFIED LEAKAGE shall be: a) leakage into primary containment collection systems, such as pump seal or valve packing leaks, that is captured and conducted to a sump or collecting tank, or b) leakage into the primary containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of the leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE.

LIMITING CONTROL ROD PATTERN (LCRP)

A LIMITING CONTROL ROD PATTERN (LCRP) shall be a pattern which results in the core being on a thermal hydraulic limit, i.e., operating on a limiting value for APLHGR, LHGR, or MCPR.

LINEAR HEAT GENERATION RATE (LHGR)

LINEAR HEAT GENERATION RATE (LHGR) shall be the heat generation per unit length of fuel rod. It is the integral of the heat flux over the heat transfer area associated with the unit length.

LOGIC SYSTEM FUNCTIONAL TEST (LSFT)

A LOGIC SYSTEM FUNCTIONAL TEST (LSFT) shall be a test of all required logic components, i.e., all required relays and contacts, trip units, solid state logic elements, etc, of a logic circuit, from sensor through and including the actuated device, to verify OPERABILITY. The LOGIC SYSTEM FUNCTIONAL TEST may be performed by any series of sequential, overlapping or total system steps such that the entire logic system is tested.

MINIMUM CRITICAL POWER RATIO (MCPR)

The MINIMUM CRITICAL POWER RATIO (MCPR) shall be the smallest CPR which exists in the core.

1.0 DEFINITIONS

OFFSITE DOSE CALCULATION MANUAL (ODCM)

The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Section 6.8 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Semi-annual Radioactive Effluent Release Reports required by Specification 6.6.

OPERABLE - OPERABILITY

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

OPERATIONAL MODE

An OPERATIONAL MODE, i.e., MODE, shall be any one inclusive combination of mode switch position and average reactor coolant temperature as specified in Table 1-2.

PHYSICS TESTS

PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and 1) described in Chapter 14 of the FSAR, 2) authorized under the provisions of 10 CFR 50.59, or 3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

PRESSURE BOUNDARY LEAKAGE shall be leakage through a non-isolable fault in a reactor coolant system component body, pipe wall or vessel wall.

1.0 DEFINITIONS

PRIMARY CONTAINMENT INTEGRITY (PCI)

PRIMARY CONTAINMENT INTEGRITY (PCI) shall exist when:

a. All primary containment penetrations required to be closed during accident conditions are either:

1) Capable of being closed by an OPERABLE primary containment automatic isolation valve system, or

2) Closed by at least one manual valve, blind flange, or deactivated automatic valve secured in its closed position, except for valves that are open under administrative control as permitted by Specification 3.7.F.

b. All primary containment equipment hatches are closed and sealed.

c. Each primary containment air lock is in compliance with the requirements of Specification 3.7.C.

d. The primary containment leakage rates are within the limits of Specification 3.7.B.

e. The suppression chamber is in compliance with the requirements of Specification 3.7.D.

f. The sealing mechanism associated with each primary containment penetration; e.g., welds, bellows or O-rings, is OPERABLE.

PROCESS CONTROL PROGRAM (PCP)

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

RATED THERMAL POWER (RTP)

RATED THERMAL POWER (RTP) shall be a total reactor core heat transfer rate to the reactor coolant of 2527 MWT.

REPORTABLE EVENT

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

1.0 DEFINITIONS

SECONDARY CONTAINMENT INTEGRITY (SCI)

SECONDARY CONTAINMENT INTEGRITY (SCI) shall exist when:

a. All secondary containment penetrations required to be closed during accident conditions are either:

   1) Capable of being closed by an OPERABLE secondary containment automatic isolation valve system, or

   2) Closed by at least one manual valve, blind flange, or deactivated automatic damper secured in its closed position, except as permitted by Specification 3.7.J.

b. All secondary containment hatches and blowout panels are closed and sealed.

c. The standby gas treatment system is in compliance with the requirements of Specification 3.7.K.

d. At least one door in each access to the secondary containment is closed.

e. The sealing mechanism associated with each secondary containment penetration; e.g., welds, bellows or O-rings, is OPERABLE.

f. The pressure within the secondary containment is less than or equal to the value required by Specification 4.7.I.1.

SHUTDOWN MARGIN (SDM)

SHUTDOWN MARGIN (SDM) shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming all control rods are fully inserted except for the single control rod of highest reactivity worth which is assumed to be fully withdrawn and the reactor is in the shutdown condition; cold, i.e. 68°F; and xenon free.

SOURCE CHECK

A SOURCE CHECK shall be the qualitative assessment of CHANNEL response when the CHANNEL sensor is exposed to a radioactive source.

STEADY STATE LINEAR HEAT GENERATION RATE (SLHGR)

The STEADY STATE LINEAR HEAT GENERATION RATE (SLHGR) shall be the limit which protects against exceeding the fuel end-of-life steady state design criteria.

THERMAL POWER

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

1.0 DEFINITIONS

TRANSIENT LINEAR HEAT GENERATION RATE (TLHGR)

The TRANSIENT LINEAR HEAT GENERATION RATE (TLHGR) shall be the limit which protects against fuel centerline melting and 1% plastic cladding strain during transient conditions throughout the life of the fuel.

TRIP SYSTEM

A TRIP SYSTEM shall be an arrangement of instrument CHANNEL trip signals and auxiliary equipment required to initiate action to accomplish a protective trip function. A TRIP SYSTEM may require one or more instrument CHANNEL trip signals related to one or more plant parameters in order to initiate TRIP SYSTEM action. Initiation of protective action may require the tripping of a single TRIP SYSTEM or the coincident tripping of two TRIP SYSTEMs.

UNIDENTIFIED LEAKAGE

UNIDENTIFIED LEAKAGE shall be all leakage in the primary containment which is not IDENTIFIED LEAKAGE.

TABLE 1-1

SURVEILLANCE FREQUENCY NOTATION

	NOTATION	FREQUENCY
1. Shift	S	At least once per 12 hours
2. Day	D	At least once per 24 hours
3. Week	W	At least once per 7 days
4. Month	M	At least once per 31 days
5. Quarter	Q	At least once per 92 days
6. Semiannual	SA	At least once per 184 days
7. Annual	A	At least once per 366 days
8. Sesquiannual	E	At least once per 18 months (550 days)
9. Startup	S/U	Prior to each reactor startup
10. Not Applicable	N.A.	Not applicable

TABLE 1-2

OPERATIONAL MODES

MODE	MODE SWITCH POSITION[f]	AVERAGE REACTOR COOLANT TEMPERATURE
1. POWER OPERATION	Run	Any temperature
2. STARTUP	Startup/Hot Standby	Any temperature
3. HOT SHUTDOWN	Shutdown[a,e]	> 212°F
4. COLD SHUTDOWN	Shutdown[a,b,e]	≤ 212°F
5. REFUELING[c]	Shutdown or Refuel[a,d]	≤ 140°F

TABLE NOTATIONS

(a) The reactor mode switch may be placed in the Run or Startup/Hot Standby position to test the switch interlock functions provided the control rods are verified to remain fully inserted by a second licensed operator or other technically qualified individual.

(b) The reactor mode switch may be placed in the Refuel position while a single control rod drive is being removed from the reactor pressure vessel per Specification 3.10.D.

(c) Fuel in the reactor vessel with one or more vessel head closure bolts less than fully tensioned or with the head removed.

(d) See Special Test Exceptions 3.12.A and 3.12.C

(e) The reactor mode switch may be placed in the Refuel position while a single control rod is being recoupled or withdrawn provided the one-rod-out interlock is OPERABLE.

(f) When there is no fuel in the reactor vessel, the reactor is considered not to be in any OPERATIONAL MODE. The reactor mode switch may then be in any position or may be inoperable.

1.0 DEFINITIONS

The following terms are defined so that uniform interpretation of these specifications may be achieved. The defined terms appear in capitalized type and shall be applicable throughout these Technical Specifications.

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

AVERAGE PLANAR EXPOSURE (APE)
The AVERAGE PLANAR EXPOSURE (APE) shall be applicable to a specific planar height and is equal to the sum of the exposure of all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle.

AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)
The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) shall be applicable to a specific planar height and is equal to the sum of the LINEAR HEAT GENERATION RATE(s) for all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle.

CHANNEL
A CHANNEL shall be an arrangement of a sensor and associated components used to evaluate plant variables and generate a single protective action signal. A CHANNEL terminates and loses its identity where single action signals are combined in a TRIP SYSTEM or logic system.

CHANNEL CALIBRATION
A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the CHANNEL output such that it responds with the necessary range and accuracy to known values of the parameter which the CHANNEL monitors. The CHANNEL CALIBRATION shall encompass the entire CHANNEL including the required sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping or total CHANNEL steps such that the entire CHANNEL is calibrated.

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

1.0 DEFINITIONS

CHANNEL FUNCTIONAL TEST

A CHANNEL FUNCTIONAL TEST shall be:

a. Analog CHANNEL(s) - the injection of a simulated signal into the CHANNEL as close to the sensor as practicable to verify OPERABILITY including required alarm and/or trip functions and CHANNEL failure trips.

b. Bistable CHANNEL(s) - the injection of a simulated signal into the sensor to verify OPERABILITY including required alarm and/or trip functions.

The CHANNEL FUNCTIONAL TEST may be performed by any series of sequential, overlapping or total CHANNEL steps such that the entire CHANNEL is tested.

CORE ALTERATION

CORE ALTERATION shall be the addition, removal, relocation or movement of fuel, sources, incore instruments or reactivity controls within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Normal movement (including replacement) of the SRMs, IRMs, TIPs, LPRMs, or special movable detectors is not considered a CORE ALTERATION. Suspension of CORE ALTERATION(s) shall not preclude completion of the movement of a component to a safe conservative position.

CORE OPERATING LIMITS REPORT (COLR)

The CORE OPERATING LIMITS REPORT (COLR) shall be the unit specific document that provides core operating limits for the current operating cycle. These cycle specific core operating limits shall be determined for each operating cycle in accordance with Specification 6.6. Plant operation within these operating limits is addressed in individual specifications.

CRITICAL POWER RATIO (CPR)

The CRITICAL POWER RATIO (CPR) shall be the ratio of that power in the assembly which is calculated by application of the applicable NRC approved critical power correlation to cause some point in the assembly to experience transition boiling, divided by the actual assembly power.

DOSE EQUIVALENT I-131

DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcurie/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors For Power and Test Reactor Sites."

FRACTION OF LIMITING POWER DENSITY (FLPD)

The FRACTION OF LIMITING POWER DENSITY (FLPD) shall be the LHGR existing at a given location divided by the specified LHGR limit for that bundle.

1.0 DEFINITIONS

FRACTION OF RATED THERMAL POWER (FRTP)
The FRACTION OF RATED THERMAL POWER (FRTP) shall be the measured THERMAL POWER divided by the RATED THERMAL POWER.

FREQUENCY NOTATION
The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1-1.

IDENTIFIED LEAKAGE
IDENTIFIED LEAKAGE shall be: a) leakage into primary containment collection systems, such as pump seal or valve packing leaks, that is captured and conducted to a sump or collecting tank, or b) leakage into the primary containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of the leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE.

LIMITING CONTROL ROD PATTERN (LCRP)
A LIMITING CONTROL ROD PATTERN (LCRP) shall be a pattern which results in the core being on a thermal hydraulic limit, i.e., operating on a limiting value for APLHGR, LHGR, or MCPR.

LINEAR HEAT GENERATION RATE (LHGR)
LINEAR HEAT GENERATION RATE (LHGR) shall be the heat generation per unit length of fuel rod. It is the integral of the heat flux over the heat transfer area associated with the unit length.

LOGIC SYSTEM FUNCTIONAL TEST (LSFT)
A LOGIC SYSTEM FUNCTIONAL TEST (LSFT) shall be a test of all required logic components, i.e., all required relays and contacts, trip units, solid state logic elements, etc, of a logic circuit, from sensor through and including the actuated device, to verify OPERABILITY. The LOGIC SYSTEM FUNCTIONAL TEST may be performed by any series of sequential, overlapping or total system steps such that the entire logic system is tested.

MAXIMUM FRACTION OF LIMITING POWER DENSITY (MFLPD)
The MAXIMUM FRACTION OF LIMITING POWER DENSITY (MFLPD) shall be the highest value of the FLPD which exists in the core.

MINIMUM CRITICAL POWER RATIO (MCPR)
The MINIMUM CRITICAL POWER RATIO (MCPR) shall be the smallest CPR which exists in the core for each class of fuel.

1.0 DEFINITIONS

OFFSITE DOSE CALCULATION MANUAL (ODCM)

The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Section 6.8 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Semi-annual Radioactive Effluent Release Reports required by Specification 6.6.

OPERABLE - OPERABILITY

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

OPERATIONAL MODE

An OPERATIONAL MODE, i.e., MODE, shall be any one inclusive combination of mode switch position and average reactor coolant temperature as specified in Table 1-2.

PHYSICS TESTS

PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and 1) described in Chapter 14 of the FSAR, 2) authorized under the provisions of 10 CFR 50.59, or 3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

PRESSURE BOUNDARY LEAKAGE shall be leakage through a non-isolable fault in a reactor coolant system component body, pipe wall or vessel wall.

1.0 DEFINITIONS

PRIMARY CONTAINMENT INTEGRITY (PCI)

PRIMARY CONTAINMENT INTEGRITY (PCI) shall exist when:

a. All primary containment penetrations required to be closed during accident conditions are either:

   1) Capable of being closed by an OPERABLE primary containment automatic isolation valve system, or

   2) Closed by at least one manual valve, blind flange, or deactivated automatic valve secured in its closed position, except for valves that are open under administrative control as permitted by Specification 3.7.F.

b. All primary containment equipment hatches are closed and sealed.

c. Each primary containment air lock is in compliance with the requirements of Specification 3.7.C.

d. The primary containment leakage rates are within the limits of Specification 3.7.B.

e. The suppression chamber is in compliance with the requirements of Specification 3.7.D.

f. The sealing mechanism associated with each primary containment penetration; e.g., welds, bellows or O-rings, is OPERABLE.

PROCESS CONTROL PROGRAM (PCP)

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

RATED THERMAL POWER (RTP)

RATED THERMAL POWER (RTP) shall be a total reactor core heat transfer rate to the reactor coolant of 2511 MWT.

REPORTABLE EVENT

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

ROD DENSITY

ROD DENSITY shall be the number of control rod notches inserted as a fraction of the total number of control rod notches. All rods fully inserted is equivalent to 100% ROD DENSITY.

1.0 DEFINITIONS

SECONDARY CONTAINMENT INTEGRITY (SCI)

SECONDARY CONTAINMENT INTEGRITY (SCI) shall exist when:

a. All secondary containment penetrations required to be closed during accident conditions are either:

   1) Capable of being closed by an OPERABLE secondary containment automatic isolation valve system, or

   2) Closed by at least one manual valve, blind flange, or deactivated automatic damper secured in its closed position, except as permitted by Specification 3.7.J.

b. All secondary containment hatches and blowout panels are closed and sealed.

c. The standby gas treatment system is in compliance with the requirements of Specification 3.7.K.

d. At least one door in each access to the secondary containment is closed.

e. The sealing mechanism associated with each secondary containment penetration; e.g., welds, bellows or O-rings, is OPERABLE.

f. The pressure within the secondary containment is less than or equal to the value required by Specification 4.7.I.1.

SHUTDOWN MARGIN (SDM)

SHUTDOWN MARGIN (SDM) shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming all control rods are fully inserted except for the single control rod of highest reactivity worth which is assumed to be fully withdrawn and the reactor is in the shutdown condition; cold, i.e. 68°F; and xenon free.

SOURCE CHECK

A SOURCE CHECK shall be the qualitative assessment of CHANNEL response when the CHANNEL sensor is exposed to a radioactive source.

THERMAL POWER

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

TRIP SYSTEM

A TRIP SYSTEM shall be an arrangement of instrument CHANNEL trip signals and auxiliary equipment required to initiate action to accomplish a protective trip function. A TRIP SYSTEM may require one or more instrument CHANNEL trip signals related to one or more plant parameters in order to initiate TRIP SYSTEM action. Initiation of protective action may require the tripping of a single TRIP SYSTEM or the coincident tripping of two TRIP SYSTEMs.

1.0 DEFINITIONS

UNIDENTIFIED LEAKAGE

UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE.

TABLE 1-1

SURVEILLANCE FREQUENCY NOTATION

	NOTATION	FREQUENCY
1. Shift	S	At least once per 12 hours
2. Day	D	At least once per 24 hours
3. Week	W	At least once per 7 days
4. Month	M	At least once per 31 days
5. Quarter	Q	At least once per 92 days
6. Semiannual	SA	At least once per 184 days
7. Annual	A	At least once per 366 days
8. Sesquiannual	E	At least once per 18 months (550 days)
9. Startup	S/U	Prior to each reactor startup
10. Not Applicable	N.A.	Not applicable

TABLE 1-2

OPERATIONAL MODES

MODE	MODE SWITCH POSITION (f)	AVERAGE REACTOR COOLANT TEMPERATURE
1. POWER OPERATION	Run	Any temperature
2. STARTUP	Startup/Hot Standby	Any temperature
3. HOT SHUTDOWN	Shutdown[a,e]	> 212°F
4. COLD SHUTDOWN	Shutdown[a,b,e]	≤ 212°F
5. REFUELING[c]	Shutdown or Refuel[a,d]	≤ 140°F

TABLE NOTATIONS

(a) The reactor mode switch may be placed in the Run or Startup/Hot Standby position to test the switch interlock functions provided the control rods are verified to remain fully inserted by a second licensed operator or other technically qualified individual.

(b) The reactor mode switch may be placed in the Refuel position while a single control rod drive is being removed from the reactor pressure vessel per Specification 3.10.D.

(c) Fuel in the reactor vessel with one or more vessel head closure bolts less than fully tensioned or with the head removed.

(d) See Special Test Exceptions 3.12.A and 3.12.B.

(e) The reactor mode switch may be placed in the Refuel position while a single control rod is being recoupled or withdrawn provided the one-rod-out interlock is OPERABLE.

(f) When there is no fuel in the reactor vessel, the reactor is considered not to be in any OPERATIONAL MODE. The reactor mode switch may then be in any position or may be inoperable.

ATTACHMENT 4

EXISTING TECHNICAL SPECIFICATIONS

Technical Specification 1.0

"DEFINITIONS"

ATTACHMENT 4

DELETION OF CURRENT TECHNICAL SPECIFICATIONS

This technical specification amendment will replace the current section 1.0, Definitions, for Dresden Unit 2 and Unit 3 Technical Specifications. The specifications are replaced in thier entirety with revised pages that combine the Unit 2 and Unit 3 specifications.

Delete the following pages:

DPR - 19	DPR - 25
1.0-1	1.0-1
1.0-2	1.0-2
1.0-3	1.0-3
1.0-4	1.0-4
1.0-5	1.0-5
1.0-6	1.0-6

ATTACHMENT 4

DELETION OF CURRENT TECHNICAL SPECIFICATIONS

This technical specification amendment will replace the current section 1.0, Definitions, for Quad Cities Unit 1 and Unit 2 Technical Specifications. The specifications are replaced in there entirety with revised pages that combine the Unit 1 and Unit 2 specifications.

Delete the following pages:

DPR - 29	DPR - 30
1.0-1	1.0-1
1.0-2	1.0-2
1.0-3	1.0-3
1.0-4	1.0-4
1.0-5	1.0-5
1.0-6	

ATTACHMENT 5

DRESDEN 2/3 DIFFERENCES

Technical Specification 1.0

"DEFINITIONS"

ATTACHMENT 5

COMPARISON OF DRESDEN UNIT 2 AND UNIT 3 TECHNICAL SPECIFICATIONS
FOR THE
IDENTIFICATION OF TECHNICAL DIFFERENCES

SECTION 1.0
"DEFINITIONS"

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.) or punctuation but rather to identify areas which the Technical Specifications are technically different.

The review of the Section 1.0, "Definitions" identified one technical difference with definition 1.0.Z, Secondary Containment Integrity. The Unit 2 Technical Specifications require the Standby Gas Treatment System to be operable whereas the Unit 3 Technical Specifications require the Standby Gas Treatment System to be in compliance with LCO 3.7.B. In the proposed specification, the definition for Secondary Containment Integrity requires the Standby Gas Treatment System to be in compliance with the specification 3.7.K in accordance with STS.

ATTACHMENT 5

QUAD CITIES 1/2 DIFFERENCES

Technical Specification 1.0

"DEFINITIONS"

ATTACHMENT 5

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

SECTION 1.0 "DEFINITIONS"

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.) or punctuation but rather to identify areas which the Technical Specifications are technically different.

The review of the Section 1.0, "Definitions" did not identify any technical differences.

ATTACHMENT 6

SIGNIFICANT HAZARDS CONSIDERATIONS AND ENVIRONMENTAL ASSESSMENT EVALUATION

Technical Specification 1.0

"DEFINITIONS"

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:

The proposed changes to the definitions are made to clarify present requirements, allow changes that have been adopted at other operating BWRs, promote consistency in understanding of the definition of terms, and to add definitions for terms used in the Dresden and Quad Cities Technical Specifications that are not currently defined.

The use of the STS and some later operating plants' version of the CORE ALTERATION definition will clearly define when this definition is applicable. Some later operating plants have revised the CORE ALTERATION definition to allow an exclusion to the definition for undervessel removal of incore instrumentation. Incorporating this change to the CORE ALTERATION definition for Dresden and Quad Cities will allow maintenance to proceed without unnecessary restrictions on plant operation and without impacting core reactivity safety while the plant is in the refuel condition. The use of STS definitions for CHANNEL CALIBRATION, CHANNEL CHECK, and CHANNEL FUNCTIONAL TEST will help to clarify the intent of the present definitions using the "Instrument" terminology. The proposed changes to the PRIMARY CONTAINMENT and SECONDARY CONTAINMENT INTEGRITY definitions will clarify present LCOs. The intent of the present definitions is that the Primary Containment Isolation Valves and Standby Gas Treatment Systems be OPERABLE pursuant to the requirements of their respective individual specifications. Present definitions could be interpreted to be more restrictive than intended and as such the changes are proposed to clarify present requirements and as such do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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The proposed change to the definition for Critical Power Ratio (CPR) follows STS guidelines and a later operating plant's version of the CPR definition. To permit the loading of a new fuel design into the Dresden or Quad Cities reactor, the change in fuel design and supporting correlations will have been previously reviewed and approved by the NRC and the limiting transients previously evaluated in the SAR will have been re-analyzed for each reload design. New core operating limits will have been generated and documented in the CORE OPERATING LIMITS REPORT (referenced in the Technical Specifications) to ensure that all safety criteria are met for all analyzed accidents and limiting transients. Therefore, this change does not involve a significant increase in the probability of consequences of any accident previously evaluated.

The changes to the definitions for FRACTION OF RATED THERMAL POWER and MINIMUM CRITICAL POWER RATIO follow STS guidelines and do not change the technical intent of the present definitions. The proposed changes to the definitions for PROCESS CONTROL PROGRAM AND OFFSITE DOSE CALCULATION MANUAL follow GL 89-01 guidelines to expand the definitions to more clearly define the content of these documents with the deletion of RETS requirements from the technical specifications. These changes do not involve a significant increase in the probability or consequences of any accident previously evaluated.

The addition of new terms to Section 1.0 provides the user of the technical specifications with easily accessible definitions that are currently accepted by other operating BWRs and are applicable to Dresden and Quad Cities. New Tables 1-1 and 1-2 allow arrangement of present Dresden and Quad Cities requirements or interpretation of requirements into an STS format for ease of use and availability. Proposed Table 1-1, "SURVEILLANCE FREQUENCY NOTATION," uses some of the present Dresden and Quad Cities interpretations of Surveillance Frequencies and does not relax or modify any existing testing intervals. Proposed Table 1-2, "OPERATIONAL MODEs," takes present requirements that are located in individual specifications and uses an STS format for arrangement of these provisions. Present temperature limits for OPERATIONAL MODEs are retained except for Refuel where the STS limit of $\leq$ 140°F is adopted. Footnotes are added to provide clarification and to allow exceptions to Mode switch position where needed to allow for necessary testing and other operations. Since the proposed addition of Tables 1-1 and 1-2 retains present operating restrictions and testing allowances or adopts proven STS guidelines that are applicable to Dresden and Quad Cities, there is no increase in the probability or consequences of an accident previously evaluated.

The applicable provisions of present Definition "Surveillance Interval" are being moved and retained in proposed Specification 4.0.B after considering implementation of Generic Letter 89-14. The present provisions of Definition

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"Shutdown," are proposed to be moved to new Table 1-2. The proposed movement of these requirements provides a more user friendly document and retains necessary limiting provisions.

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

The proposed changes leave intact present operating philosophy and only implement new provisions where necessary to clarify and ensure that present allowances are understood and maintained. The proposed exception to the CORE ALTERATION definition will allow replacement of incore instruments without considering this an alteration of the core. Due to the small amount of fissionable material in these instruments, their movement cannot create the possibility of a new or different kind of accident from any previously evaluated for nuclear safety.

The change from the "Instrument" philosophy to the "CHANNEL" philosophy for calibrations, checks and functional tests provides clarification of present wording and intent. The STS "CHANNEL" philosophy is being used at Dresden and Quad Cities in many present applications and this technical specification change will ensure its consistent use for applicable testing functions.

The proposed changes to the PRIMARY CONTAINMENT and SECONDARY CONTAINMENT INTEGRITY definitions are clarifications of intent with respect to OPERABILITY provisions for Primary Containment Isolation Valves and Standby Gas Treatment Systems. Primary Containment and SECONDARY CONTAINMENT INTEGRITY requirements are considered to be maintained while equipment and systems are in the ACTION statements of Specifications 3.7.C, 3.7.F and 3.7.K. The individual equipment and system specifications contain allowed outage provisions to ensure that OPERABILITY is maintained within defined time limits, which are considered to be sufficiently short in duration, such that impact is minimal to PRIMARY or SECONDARY CONTAINMENT INTEGRITY considerations. The proposed changes do not change system OPERABILITY requirements and as such do not create the possibility of a new or different kind of accident from any previously evaluated.

Use of a more generic reference "applicable NRC-approved critical power correlation" in the CRITICAL POWER RATIO definition in place of a reload specific correlation like "GEXL" will not change the present intent of the definition but only preclude the necessity to revise the CRITICAL POWER RATIO definition every time there are minor changes in the fuel manufacturer's critical power correlations to support their new fuel design. Provided that the changes to the critical power correlation are reviewed and approved by the NRC, no new or different accident, from any previously evaluated, is created by this broader definition. Therefore, this change cannot create the possibility of a new or different kind of accident from any previously evaluated.

The changes to the definitions for FRACTION OF RATED THERMAL POWER and MINIMUM CRITICAL POWER RATIO do not change present intent and are made to clarify present requirements. The changes to the definitions for PCP and ODCM are expansions of present provisions in order to implement the provisions of GL 89-01. Due to the nature of these changes, they cannot create the possibility of a new or different kind of accident.

The addition of new definitions to the technical specifications is an enhancement to present provisions. STS guidelines are used for the new definitions and have been evaluated and found to be in agreement with present usage at Dresden and Quad Cities. New Tables 1-1 and 1-2 follow an STS format for implementing present Dresden and Quad Cities Surveillance Frequencies and OPERATIONAL MODEs. Present OPERATIONAL MODE restrictions on reactor coolant temperature are retained for OPERATIONAL MODEs 1, 2, 3, and 4; and, are changed to STS guidelines for OPERATIONAL MODE 5. Present reactor mode switch position restrictions are retained by including necessary notes to allow testing and other operations. Since either present provisions are retained or present interpretation of requirements are maintained, the changes do not create the possibility of a new or different kind of accident from any previously evaluated.

Involve a significant reduction in the margin of safety because:

The proposed changes to the definitions provide clarifications, implement proven changes from operating BWRs that are applicable at Dresden and Quad Cities, and include present provisions and interpretations presented in STS format. Present margins of safety are retained and improved by clarifying requirements that are subject to interpretation or are not presented in an easy to understand format.

The proposed change to the CORE ALTERATION definition does not affect nuclear safety since replacement of incore instrumentation has little or no measurable impact on core reactivity. The change from the "Instrument" to the "CHANNEL" philosophy for calibration, checks, and functional tests provides clarification of present intent. Clarifications to the definitions for PRIMARY and SECONDARY CONTAINMENT INTEGRITY are made to prevent misinterpretation of intent of the present requirements and do not reduce any margin of safety. The proposed revision to the definition of CRITICAL POWER RATIO will merely redefine, in broader terms, the definition of CRITICAL POWER RATIO and will not cause a change in any margin of safety. Each fuel reload analyses will continue to ensure that the fuel system design, nuclear design, thermal/hydraulic design and the conclusions of the original core analysis remain valid for the accidents and limiting transients previously evaluated in the SAR. The changes to the definitions for FRACTION OF RATED THERMAL POWER and MINIMUM CRITICAL POWER RATIO are clarifications of present requirements that do not change present technical intent. The changes to the definitions for PCP and ODCM more clearly define the contents of these documents with the implementation of GL 89-01 provisions. As such these changes cannot reduce any margin of safety.

The new definitions added to Section 1.0 apply to terms in current use in the Dresden and Quad Cities Technical Specifications and this addition improves understanding of requirements. New Tables 1-1 and 1-2 follow the STS in format with notations and OPERATIONAL MODEs based on present Dresden and Quad Cities Technical Specification requirements, interpretation of requirements, or STS guidelines that are applicable to Dresden and Quad Cities. Table 1-2 notes follow present Dresden and Quad Cities allowances or interpretation of allowances and follow later operating plants and STS guidelines. Since the proposed changes implement present Dresden and Quad Cities allowances in an STS format and follow proven allowances at other operating plants that are acceptable for use at Dresden and Quad Cities, there is no reduction in any 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 1

EXECUTIVE SUMMARY

Technical Specification 3/4.0

"<u>APPLICABILITY</u>"

EXECUTIVE SUMMARY

The Dresden Technical Specification Upgrade Program (TSUP) was conceptualized in response to lessons learned from the Diagnostic Evaluation Team inspection and the frequent need for Technical Specification interpretations. A comparison study of the Standard Technical Specification (STS), later operating plant's Technical Specifications provisions and Quad Cities Technical Specifications was performed prior to the Dresden TSUP effort. The study identified potential improvements in clarifying requirements and requirements which are no longer consistent with current industry practices. The Dresden TSUP will enhance the Quad Cities TSUP currently under review by the NRC. As a result of the inconsistencies in the Quad Cities submittal compared to the Standard Technical Specifications (STS), Dresden's submittal will more closely follow the provisions of STS and in conjunction, Quad Cities will amend their submittals so that Quad Cities and Dresden are identical within equipment and plant design. The format for the Dresden TSUP will remain as a two column layout for human factors considerations. Additionally, chapter organizations will remain unchanged.

The TSUP is not intended to be a complete adoption for the STS. Overall, the Dresden custom Technical Specifications provide for the safe operation of the plant and therefore, only an upgrade is deemed necessary.

In response to an NRC recommendation, Quad Cities combined the Unit 1 and Unit 2 Technical Specifications into one document. The Dresden Unit 2 and Unit 3 Technical Specifications will also be combined into one document. To accomplish the combination of the Units' Technical Specification, a comparison of the Unit 2 and Unit 3 Technical Specification was performed to identify any technical differences. The technical differences are identified in the proposed amendment package for each section.

The TSUP was identified as a station top priority and is currently contained in the Dresden Management Action Plan (DMAP). The TSUP goal is to provide a better tool to station personnel to implement their responsibilities and to ensure Dresden Station is operated in accordance with current industry practices. The improved Technical Specifications provide for enhanced operation of the plant. The program improves the operator's ability to use the Technical Specifications by more clearly defining the Limiting Conditions for Operation and required actions. The most significant improvement to the specifications is the addition of equipment operability requirements during shutdown conditions.

EXECUTIVE SUMMARY

(continued)

PROPOSED CHANGES TO TECHNICAL SPECIFICATION
SECTION 3.0/4.0, "APPLICABILITY"

Present section 3.0, "Limiting Condition for Operation" contains three specifications. Limiting Conditions for Operation (LCO) 3.0.A addresses the action to be taken if a LCO cannot be satisfied. LCO 3.0.B delineates the additional conditions which must be satisfied to permit continued operation when a normal or emergency power source is not operable. LCO 3.0.C subscribes that 3.0.A or 3.0.B are not applicable in the Refuel or Shutdown mode.

The proposed amendment retains these provisions and adds requirements not currently delineated in the specifications. The general content of these new provisions are accepted as standard operating practice at later boiling water reactor plants and are in conformance with the Standard Technical Specifications and Generic Letters 87-09 and 89-14.

The proposed amendment to specification 3.0/4.0 are intended to accomplish the following:

- Provides direction regarding LCO compliance during specific reactor modes and the associated action requirements upon failure to meet a LCO.
- Defines noncompliance with a specification and the required actions associated with restoration of a LCO.
- Defines the necessary actions for those circumstances not directly provided for in the action statement of a specification.
- Provides guidance to define when entry into an operational mode or other specified condition is allowed if the plant is operating under the action provisions of a LCO.
- Specifies when surveillance requirements shall be met and the time interval allowed for performing surveillance requirements (per current Definition 1.0.CC) and, defines a failure to satisfy a surveillance requirement.
- Specifies that entry into OPERATIONAL MODEs or other specified conditions shall not be made unless the surveillance requirement associated with the LCO have been performed within the applicable surveillance interval or as otherwise specified.
- Relocates Inservice Inspection and Inservice Testing requirements of ASME Code Class 1, 2, and 3 components from specification 4.6.F.

ATTACHMENT 2

DESCRIPTION OF CHANGES

Technical Specification 3/4.0

"APPLICABILITY"

ATTACHMENT 2

DESCRIPTION OF PROPOSED AMENDMENT

CURRENT REQUIREMENTS

Section 3.0, Limiting Conditions for Operation contains three specifications. LCO 3.0.A addresses the action to be taken if a LCO cannot be satisfied because of circumstances in excess of those listed in the specification. LCO 3.0.B delineates what additional conditions must be satisfied to permit operation to continue, consistent with the LCOs for power sources, when a normal or emergency power source is not operable. LCO 3.0.C states that LCOs 3.0.A and 3.0.B are not applicable in the refueling or cold shutdown modes. The present 3.0 specifications do not contain all normally accepted upper level provisions governing the use and applicability of surveillance requirements for the individual specifications. The present 3.0 specifications are lacking in content and do not contain upper level provisions governing the use and applicability of surveillance requirements for the individual specifications.

PROPOSED AMENDMENT

The proposed amendment request contains changes to Dresden Units 2 and 3 and Quad Cities Units 1 and 2 Technical Specifications (TS) Section 3/4.0 on Applicability. The changes constitute a complete adoption of the Standard Technical Specifications (STS) 3.0/4.0 section, the changes proposed by the NRC in Generic Letter 87-09 and the change in surveillance intervals presented in Generic Letter 89-14, with one exception. The exception is the shutdown times and modes contained in section 3.0.C. The following discussions provide details on each of the proposed sections contained in Section 3/4.0.

Section 3.0.A: This is a new section which is a complete adoption of the STS section 3.0.1. This section provides direction such that a LCO must be complied with during specified reactor modes or other specified conditions and upon failure to meet a LCO, the associated ACTION requirements must be met.

Section 3.0.B: This is a new section which is a complete adoption of the STS section 3.0.2. This section defines when noncompliance with a specification exists. Noncompliance exists when the LCO and ACTION provisions are not met within the specified time intervals. Also 3.0.B specifies that if the LCO is restored, completion of the ACTION requirements is not required thus removing the need for testing, plant shutdown or other ACTION provisions in progress.

Section 3.0.C: This is a new replacement for the current 3.0.A to clearly define the actions necessary to be taken for those circumstances not directly provided for in the ACTION statements of a specification. Changes to the present specification will improve this by requiring ACTION within one hour to place the unit in a reactor mode or other condition in which the specification does not apply. The provisions will ensure that steps are being taken in a timely manner to place the plant in a reactor mode or other condition where the inoperable equipment is not required. Any exceptions to the proposed provisions must be stated in the individual specifications and thus must have prior NRC approval. Retained from the present specifications are the provisions to be in at least HOT SHUTDOWN within the next 12 hours and at least COLD SHUTDOWN within the subsequent 24 hours, after the one hour allowance. The 12/24 allowance differs from the STS (6/6/24) but is more applicable to the mode of operation for Quad Cities and Dresden. Since the STS was written with the 6/6/24 shutdown requirements, operating restrictions and practices have increased the time required to achieve an orderly shutdown such that 6 hour time frames from full power to hot standby and from hot standby to hot shutdown are becoming increasingly difficult to attain. Rather than place such unnecessary additional constraints on the operators at Dresden and Quad Cities, the current (and essentially equivalent) requirements are retained. This section also contains the provisions of current specification 3.0.C, which is in accordance with the STS.

Section 3.0.D: This is a new section which is a complete adoption of STS section 3.0.4, with one minor clarification, as set forth in Generic Letter 87-09. An individual analysis of each specific LCO ACTION statement and its applicability per the requirements of Generic Letter 87-09 will be explained with each subsequent transmittal of proposed changes to TS sections under the Technical Specification Upgrade Program. Section 3.0.D provides guidance when entry into an OPERATIONAL MODE or other specified condition is allowed if the plant is operating with systems or equipment under the ACTION provisions of a LCO. Entry into an OPERATIONAL MODE or other specified condition is allowed in accordance with the ACTION requirements when conformance to the ACTION requirements permits continued operation of the facility for an unlimited period of time. A minor revision has been proposed to the wording of 3.0.D that differs from the Generic Letter 87-09 proposed language. The Generic Letter proposed language indicates that "Entry into an OPERATIONAL MODE or other specified condition shall not be made when the conditions for the LCO are not met and the associated ACTION requires a shutdown if they are not met within a specified time interval. This phrase could be interpreted to imply that the applicable OPERATIONAL MODE or other specified condition could be entered while the LCO is not met and the ACTION requires that the plant be placed in a non-applicable OPERATIONAL MODE within a specified time frame but does not require a shutdown. For example, STS 3.2.1 is applicable only in Mode 1 above 25% power. The ACTION requires restoration of operation within the LCO limits or a reduction of power to less than 25% within 4 hours, but does not require a shutdown. Therefore, a strict interpretation would allow entry into MODE 1 and an increase of power above 25% while operating in accordance with this ACTION. However, Enclosure 1 to Generic Letter 87-09 (last sentence of paragraph 2 under problem #1, BACKGROUND) provides additional clarification of the intended use of the term "shutdown." It states "...action must be taken to shutdown the facility by placing it in a mode or condition of operation in which the LCO does not apply." This clarification has been incorporated into the proposed 3.0.D and into the Bases.

Section 4.0.A: This is a new section which is a complete adoption of STS section 4.0.1. Section 4.0.A specifies when surveillance requirements shall be met. Unless otherwise stated in an individual specification, the surveillance requirements shall be met during the reactor mode or other specified conditions for the LCO.

Section 4.0.B: This is a new section that combines current definition 1.0.CC and STS section 4.0.2 as modified by Generic Letter 89-14. Section 4.0.B specifies the time interval allowed for performing surveillance requirements. The current definition "Surveillance Interval" that specifies a maximum allowable extension of 25% of the surveillance interval is included in section 4.0.B in accordance with the STS specification. The Generic Letter recommended the removal of the 3.25 limit for maximum extension of consecutive surveillances. The NRC concluded in this letter that the removal of the 3.25 limit results in a greater benefit to safety than limiting the use of the 25% allowance to extend surveillance intervals. Commonwealth Edison has evaluated the provisions of this Generic Letter and concluded that incorporation at Dresden and Quad Cities will represent an improvement over the present requirements by allowing for more operational flexibility when scheduling surveillances. This flexibility will allow surveillances to be performed when plant conditions are appropriate for the testing. The proposed wording for the specification was taken directly from the Generic Letter.

Section 4.0.C: This is a new section which is a complete adoption of STS section 4.0.3. and Generic Letter 87-09. This section defines the meaning of failure to satisfy a surveillance requirement. This failure results in a failure to meet the OPERABILITY requirements for a LCO. The time limits of the action requirements are initiated at the time it is identified that a surveillance requirement has not been performed within the maximum allowed surveillance interval. The ACTION may be delayed for up to 24 hours to permit completion of the surveillance when the allowable outage time limits of the action requirements are less than 24 hours. Any exceptions to these provisions must be stated in the individual specification and thus must have prior NRC approval.

Section 4.0.D: This is a new section that is a complete adoption of STS section 4.0.4 and Generic Letter 87-09. This section specifies that entry into OPERATIONAL MODEs or other specified conditions shall not be made unless the surveillance requirements associated with the LCO have been performed within the applicable surveillance interval or as otherwise specified. This specification helps to e.sure equipment operability when required by the Technical Specification Applicability statement. This requirement does not prevent passage through or to OPERATIONAL MODEs as required to comply with action requirements. Exceptions to these requirements must be stated in the individual specifications.

Section 4.0.E: This is a new section that combines the requirements of current section 4.6.F, STS section 4.0.5 and the requirements in Generic Letter 88-01. This section establishes the requirement that Inservice Inspection of ASME Code Class 1, 2, and 3 components and Inservice Testing of ASME Code Class 1, 2, and 3 pumps and valves shall be performed in accordance with a periodically updated version of Section XI of the ASME Boiler and Press.re Vessel Code and Addenda as required by 10 CFR 50.55a. Under the terms of this specification, the more restrictive requirements of the Technical Specification take precedence over the ASME Boiler and Pressure Vessel Code and applicable Addenda.

Bases 3/4.0: These Bases represent a complete adoption of the STS Bases for Section 3/4.0 as modified by Generic Letters 87-09, 89-14 and 91-04.

Summary

The NRC issued Generic Letter 87-09 outlining recent initiatives undertaken by the NRC and the nuclear industry to improve Technical Specifications. The generic letter provided guidance for three specific problems encountered with the general requirements on the applicability of LCO and surveillance requirements in sections 3.0 and 4.0. The generic letter included the NRC modified version of section 3.0 and 4.0 of the BWR STS, and provided the NRCs updated version of the BWR STS Bases for these sections. The generic letter stated that the NRC staff has concluded that these modifications will improve the Technical Specifications for all plants, and encouraged licensees to propose changes to their Technical specifications consistent with the generic letter guidance. Commonwealth Edison has reviewed the generic letter with its enclosures and concluded that the proposed modifications as proposed in this amendment are an improvement over the present Dresden and Quad Cities Technical Specifications. As such the proposed amendment represents a complete adoption of the wording and requirements outlined in the STS as modified by the generic letters with the one exception discussed in section 3.0.D.

ATTACHMENT 3

PROPOSED TECHNICAL SPECIFICATIONS

Technical Specification 3/4.0

"APPLICABILITY"

3.0 - LIMITING CONDITIONS FOR OPERATION

A. Compliance with the Limiting Conditions for Operation contained in the succeeding Specifications is required during the OPERATIONAL MODE(s) or other conditions specified therein; except that upon failure to meet the Limiting Conditions for Operation, the associated ACTION requirements shall be met.

B. Noncompliance with a Specification shall exist when the requirements of the Limiting Condition for Operation and associated ACTION requirements are not met within the specified time intervals. If the Limiting Condition for Operation is restored prior to expiration of the specified time intervals, completion of the ACTION requirements is not required.

C. When a Limiting Condition for Operation is not met, except as provided in the associated ACTION requirements, within one hour ACTION shall be initiated to place the unit in an OPERATIONAL MODE in which the Specification does not apply by placing it, as applicable, in:

1. At least HOT SHUTDOWN within the next 12 hours, and
2. At least COLD SHUTDOWN within the subsequent 24 hours.

Where corrective measures are completed that permit operation under the ACTION requirements, the ACTION may be taken in accordance with the specified time limits as measured from the time of failure to meet the Limiting Condition for Operation. Exceptions to these requirements are stated in the individual Specifications.

This Specification is not applicable in OPERATIONAL MODE 4 or 5.

D. Entry into an OPERATIONAL MODE or other specified condition shall not be made when the conditions for the Limiting Conditions for Operation are not met and the associated ACTION requires placing the plant in an OPERATIONAL MODE or other specified condition of operation in which the Limiting Condition for Operation does not apply if they are not met within a specified time interval. Entry into an OPERATIONAL MODE or other specified condition may be made in accordance with the ACTION requirements when conformance to them permits continued operation of the facility for an unlimited period of time. This provision shall not prevent passage through or to OPERATIONAL MODE(s) as required to comply with ACTION requirements. Exceptions to these requirements are stated in the individual Specifications.

4.0 - SURVEILLANCE REQUIREMENTS

A. Surveillance Requirements shall be met during the reactor OPERATIONAL MODE(s) or other conditions specified for individual Limiting Conditions for Operation unless otherwise stated in an individual Surveillance Requirement.

B. Each Surveillance Requirement shall be performed within the specified surveillance interval with a maximum allowable extension not to exceed 25 percent of the surveillance interval.

C. Failure to perform a Surveillance Requirement within the allowed surveillance interval, defined by Specification 4.0.B, shall constitute noncompliance with the OPERABILITY requirements for a Limiting Condition for Operation. The time limits of the ACTION requirements are applicable at the time it is identified that a Surveillance Requirement has not been performed. The ACTION requirements may be delayed for up to 24 hours to permit the completion of the surveillance when the allowable outage time limits of the ACTION requirements are less than 24 hours. Surveillance requirements do not have to be performed on inoperable equipment.

D. Entry into an OPERATIONAL MODE or other specified applicable condition shall not be made unless the Surveillance Requirement(s) associated with the Limiting Condition for Operation have been performed within the applicable surveillance interval or as otherwise specified. This provision shall not prevent passage through or to OPERATIONAL MODE(s) as required to comply with ACTION requirements.

E. Surveillance Requirements for inservice inspection and testing of ASME Code Class 1, 2, and 3 components shall be applicable as follows:

1. Inservice Inspection of ASME Code Class 1, 2, and 3 components and inservice testing of ASME Code Class 1, 2, and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR Part 50, Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 50, Section 50.55a(g)(6)(i).

4.0 - SURVEILLANCE REQUIREMENTS

2. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice inspection and testing activities required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these Technical Specifications:

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice inspection and testing activities	Required Frequencies for performing inservice inspection and testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days

3. The provisions of Specification 4.0.B are applicable to the above required frequencies for performing inservice inspection and testing activities.

4. Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements.

5. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any Technical Specification.

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

BASES

Specifications 3.0.A through 3.0.D establish the general requirements applicable to Limiting Conditions for Operation. These requirements are based on the requirements for Limiting Conditions for Operation stated in the Code of Federal Regulations, 10 CFR 50.36(c)(2):

> "Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specification until the condition can be met."

<u>Specification 3.0.A</u> establishes the Applicability statement within each individual specification as the requirement for when (i.e., in which OPERATIONAL MODE(s) or other specified conditions) conformance to the Limiting Conditions for Operation is required for safe operation of the facility. The ACTION requirements establish those remedial measures that must be taken within specified time limits when the requirements of a Limiting Condition for Operation are not met. It is not intended that the shutdown ACTION requirements be used as an operational convenience which permits (routine) voluntary removal of a system(s) or component(s) from service in lieu of other alternatives that would not result in redundant systems or components being inoperable.

There are two basic types of ACTION requirements. The first specifies the remedial measures that permit continued operation of the facility which is not further restricted by the time limits of the ACTION requirements. In this case, conformance to the ACTION requirements provides an acceptable level of safety for unlimited continued operation as long as the ACTION requirements continue to be met. The second type of ACTION requirement specifies a time limit in which conformance to the conditions of the Limiting Condition for Operation must be met. This time limit is the allowable outage time to restore an inoperable system or component to OPERABLE status or for restoring parameters within specified limits. If these ACTION(s) are not completed within the allowable outage time limits, a shutdown is required to place the facility in a reactor OPERATIONAL MODE or other specified condition in which the specification no longer applies.

The specified time limits of the ACTION requirements are applicable from the point in time it is identified that a Limiting Condition for Operation is not met. The time limits of the ACTION requirements are also applicable when a system or component is removed from service for surveillance testing or investigation of operational problems. Individual specifications may include a specified time limit for the completion of a Surveillance Requirement when equipment is removed from service. In this case, the allowable outage time limits of the ACTION requirements are applicable when this limit expires if the surveillance has not been completed. When a shutdown is required to comply with ACTION requirements, the plant may have entered an OPERATIONAL MODE in which a new specification becomes applicable. In this case, the time limits of the ACTION requirements would apply from the point in time that the new specification becomes applicable if the requirements of the Limiting Condition for Operation are not met.

<u>Specification 3.0.B</u> establishes that noncompliance with a specification exists when the requirements of the Limiting Condition for Operation are not met and the associated ACTION

requirements have not been implemented within the specified time interval. The purpose of this specification is to clarify that (1) implementation of the ACTION requirement within the specified time interval constitutes compliance with a specification and (2) completion of the remedial measures of the ACTION requirements is not required when compliance with a Limiting Condition for Operation is restored within the time interval specified in the associated ACTION requirements.

Specification 3.0.C establishes the shutdown ACTION requirements that must be implemented when a Limiting Condition for Operation is not met and the condition is not specifically addressed by the associated ACTION requirements. The purpose of this specification is to delineate the time limits for placing the unit in a safe shutdown condition when plant operation cannot be maintained within the limits for safe operation defined by the Limiting Condition for Operation and its ACTION requirements. It is not intended to be used as an operational convenience which permits (routine) voluntary removal of redundant systems or components from service in lieu of other alternatives that would not result in redundant systems or components being inoperable. One hour is allowed to prepare for an orderly shutdown before initiating a change in plant operation. This time permits the operator to coordinate the reduction in electrical generation with the load dispatcher to ensure the stability and availability of the electrical grid. The time limits specified to reach lower MODE(s) of operation permit the shutdown to proceed in a controlled and orderly manner that is well within the specified maximum cooldown rate and within the cooldown capabilities of the facility assuming only the minimum required equipment is OPERABLE. This reduces thermal stresses on components of the primary coolant system and the potential for a plant transient that could challenge safety systems under conditions for which this specification applies.

If remedial measures permitting limited continued operation of the facility under the provisions of the ACTION requirements are completed, the shutdown may be terminated. The time limits of the ACTION requirements are applicable from the point in time there was a failure to meet a Limiting Condition for Operation. Therefore, the shutdown may be terminated if the ACTION requirements have been met or the time limits of the ACTION requirements have not expired, thus providing an allowance for the completion of the required ACTION(s).

The time limits of Specification 3.0.C allow 37 hours for the plant to be in COLD SHUTDOWN when a shutdown is required during POWER OPERATION. If the plant is in a lower MODE of operation when a shutdown is required, the time limit for reaching the next lower MODE applies. However, if a lower MODE of operation is reached in less time than allowed, the total allowable time to reach COLD SHUTDOWN, or other OPERATIONAL MODE, is not reduced. For example, if HOT SHUTDOWN is reached in 10 hours, the time allowed to reach COLD SHUTDOWN is the next 27 hours because the total time to reach COLD SHUTDOWN is not reduced from the allowable limit of 37 hours. Therefore, if remedial measures are completed that would permit a return to POWER OPERATION, a penalty is not incurred by having to reach a lower MODE of operation in less than the total time allowed.

The same principle applies with regard to the allowable outage time limits of the ACTION requirements, if compliance with the ACTION requirements for one specification results in entry into an OPERATIONAL MODE or condition of operation for another specification in which the requirements of the Limiting Condition for Operation are not met. If the new specification becomes applicable in less time than specified, the difference may be added to the allowable outage time limits of the second specification. However, the allowable outage time limits of ACTION requirements for a higher MODE of operation may not be used to extend the allowable outage time that is applicable when a Limiting Condition for Operation is not met in a lower MODE of operation.

The shutdown requirements of Specification 3.0.C do not apply in MODES 4 or 5, because the ACTION requirements of individual specifications define the remedial measures to be taken.

Specification 3.0.D establishes limitations on a change in OPERATIONAL MODE(s) when a Limiting Condition for Operation is not met. It precludes placing the facility in a higher MODE of operation when the requirements for a Limiting Condition for Operation are not met and continued noncompliance to these conditions would result in placing the plant in an OPERATIONAL MODE or other specified condition of operation in which the Limiting Condition for Operation does not apply to comply with the ACTION requirements if a change in MODE(s) were permitted. The purpose of this specification is to ensure that facility operation is not initiated or that higher MODE(s) of operation or other specified conditions are not entered when corrective ACTION is being taken to obtain compliance with a specification by restoring equipment to OPERABLE status or parameters to specified limits. Compliance with ACTION requirements that permit continued operation of the facility for an unlimited period of time provides an acceptable level of safety for continued operation without regard to the status of the plant before or after a change in OPERATIONAL MODE(s). Therefore, in this case, entry into an OPERATIONAL MODE or other specified condition may be made in accordance with the provisions of the ACTION requirements. The provisions of this specification should not, however, be interpreted as endorsing the failure to exercise good practice in restoring systems or components to OPERABLE status before plant startup.

When a shutdown is required to comply with ACTION requirements, the provisions of Specification 3.0.D do not apply because they would delay placing the facility in a lower MODE of operation.

BASES

Specifications 4.0.A through 4.0.E establish the general requirements applicable to Surveillance Requirements. These requirements are based on the Surveillance Requirements stated in the Code of Federal Regulations, 10 CFR 50.36(c)(3):

> "Surveillance requirements are requirements relating to test, calibration, or inspection to ensure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions of operation will be met."

Specification 4.0.A establishes the requirement that surveillances must be performed during the OPERATIONAL MODE(s) or other conditions for which the requirements of the Limiting Condition for Operation apply unless otherwise stated in an individual Surveillance Requirement. The purpose of this specification is to ensure that surveillances are performed to verify the operational status of systems and components and that parameters are within specified limits to ensure safe operation of the facility when the plant is in a reactor OPERATIONAL MODE or other specified condition for which the individual Limiting Condition for Operations are applicable. Surveillance Requirements do not have to be performed when the facility is in an OPERATIONAL MODE for which the requirements of the associated Limiting Condition for Operation do not apply unless otherwise specified. The Surveillance Requirements associated with a Special Test Exception are only applicable when the Special Test Exception is used as an allowable exception to the requirements of a specification.

Specification 4.0.B establishes the limit for which the specified time interval for Surveillance Requirements may be extended. It permits an allowable extension of the normal surveillance interval to facilitate surveillance scheduling and consideration of plant operating conditions that may not be suitable for conducting the surveillance; e.g., transient conditions or other ongoing surveillance or maintenance activities. It also provides flexibility to accommodate the length of a fuel cycle for surveillances that are specified with and 18 month surveillance interval. It is not intended that this provision be used repeatedly as a convenience to extend surveillance intervals beyond that specified for surveillances that are not performed during refueling outages. Likewise, it is not the intent that refueling outage surveillances be performed during power operation unless it is consistent with safe plant operation. The limitation of Specification 4.0.B is based on engineering judgment and the recognition that the most probable result of any particular surveillance being performed is the verification of conformance with the Surveillance Requirements. This provision is sufficient to ensure that the reliability ensured through surveillance activities is not significantly degraded beyond that obtained from the specified surveillance interval.

Specification 4.0.C establishes that the failure to satisfy a Surveillance Requirement within the allowed surveillance interval, defined by the provisions of Specification 4.0.B, is a condition that constitutes a failure to meet the OPERABILITY requirements for a Limiting Condition for Operation. Under the provisions of this specification, systems and components are assumed to be OPERABLE when Surveillance Requirements have been satisfactorily performed within the specified time interval. However, nothing in this provision is to be construed as implying that systems or components are OPERABLE when they are found or known to be inoperable although still meeting the Surveillance Requirements. This specification also clarifies that the ACTION requirements are applicable when Surveillance Requirements have not been completed within the allowed surveillance interval and that the time limits of the ACTION requirements apply from the point in time it is identified that a surveillance has not been performed and not at the time that the allowed surveillance interval was exceeded. Completion of the Surveillance Requirement within the allowable outage time limits of the ACTION requirements restores compliance with the requirements of Specification 4.0.C. However, this does not negate the fact that the failure to have performed the surveillance within the allowed surveillance interval, defined by the provisions of Specification 4.0.B, was a violation of the OPERABILITY requirements for a Limiting Condition for Operation that is subject to enforcement action. The failure to perform a surveillance within the provisions of Specification 4.0.B is a violation of a Technical Specification requirement and is, therefore, a reportable event under the requirements of 10 CFR 50.73(a)(2)(i)(B) because it is a condition prohibited by the plant's Technical Specifications.

If the allowable outage time limits of the ACTION requirements are less than 24 hours or a shutdown is required to comply with ACTION requirements (e.g., in Specification 3.0.C), a 24-hour allowance is provided to permit a delay in implementing the ACTION requirements. This provides an adequate time limit to complete Surveillance Requirements that have not been performed. The purpose of this allowance is to permit the completion of a surveillance before a shutdown would be required to comply with ACTION requirements or before other remedial measures would be required that may preclude the completion of a surveillance. The basis for this allowance includes consideration for plant conditions, adequate planning, availability of personnel, the time required to perform the surveillance, and the safety significance of the delay in completing the required surveillance. This provision also provides a time limit for the completion of Surveillance Requirements that become applicable as a consequence of MODE changes imposed by ACTION requirements and for completing Surveillance Requirements that are applicable when an exception to the requirements of Specification 4.0.D is allowed. If a surveillance is not completed within the 24-hour allowance, the time limits of the ACTION requirements are applicable at that time. When a surveillance is performed within the 24-hour allowance and the Surveillance Requirements are not met, the time limits of the ACTION requirements are applicable at the time that the surveillance is terminated.

Surveillance Requirements do not have to be performed on inoperable equipment because the ACTION requirements define the remedial measures that apply. However, the Surveillance Requirements have to be met to demonstrate that inoperable equipment has been restored to OPERABLE status.

Specification 4.0.D establishes the requirement that all applicable surveillances must be met before entry into an OPERATIONAL MODE or other condition of operation specified in the Applicability statement. The purpose of this specification is to ensure that system and component OPERABILITY requirements or parameter limits are met before entry into an OPERATIONAL MODE or other specified condition for which these systems and components ensure safe operation of the facility. This provision applies to changes in OPERATIONAL MODE(s) or other specified conditions associated with plant shutdown as well as startup.

Under the provisions of this specification, the applicable Surveillance Requirements must be performed within the specified surveillance interval to assure that the Limiting Conditions for Operation are met during initial plant startup or following a plant outage

When a shutdown is required to comply with ACTION statements, the provisions of Specification 4.0.D do not apply because this would delay placing the facility in a lower MODE of operation.

Specification 4.0.E establishes the requirement that inservice inspection of ASME Code Class 1, 2, and 3 components and inservice testing of ASME Code Class 1, 2, and 3 pumps and valves shall be performed in accordance with a periodically updated version of Section XI of the ASME Boiler and Pressure Vessel Code and Addenda as required by 10 CFR 50.55a. These requirements apply except when relief has been provided in writing by the Commission.

This specification includes a clarification of the frequencies for performing the inservice inspection and testing activities required by Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda. This clarification is provided to ensure consistency in surveillance intervals throughout the Technical Specifications and to remove any ambiguities relative to the frequencies for performing the required inservice inspection and testing activities.

Under the terms of this specification, the more restrictive requirements of the Technical Specifications take precedence over the ASME Boiler and Pressure Vessel Code and applicable Addenda. The requirements of Specification 4.0.D to perform surveillance requirements before entry into an OPERATIONAL MODE or other specified condition takes precedence over the ASME Boiler and Pressure Vessel Code provision that allows pumps and valves to be tested up to one week after return to normal operation. The Technical Specification definition of OPERABLE does not allow a grace period before a component, which is not capable of performing its specified function, is declared inoperable and takes precedence over the ASME Boiler and Pressure Vessel Code provision that allows a valve to be incapable of performing its specified function for up to 24 hours before being declared inoperable.

3.0 - LIMITING CONDITIONS FOR OPERATION

A. Compliance with the Limiting Conditions for Operation contained in the succeeding Specifications is required during the OPERATIONAL MODE(s) or other conditions specified therein; except that upon failure to meet the Limiting Conditions for Operation, the associated ACTION requirements shall be met.

B. Noncompliance with a Specification shall exist when the requirements of the Limiting Condition for Operation and associated ACTION requirements are not met within the specified time intervals. If the Limiting Condition for Operation is restored prior to expiration of the specified time intervals, completion of the ACTION requirements is not required.

C. When a Limiting Condition for Operation is not met, except as provided in the associated ACTION requirements, within one hour ACTION shall be initiated to place the unit in an OPERATIONAL MODE in which the Specification does not apply by placing it, as applicable, in:

   1. At least HOT SHUTDOWN within the next 12 hours, and

   2. At least COLD SHUTDOWN within the subsequent 24 hours.

   Where corrective measures are completed that permit operation under the ACTION requirements, the ACTION may be taken in accordance with the specified time limits as measured from the time of failure to meet the Limiting Condition for Operation. Exceptions to these requirements are stated in the individual Specifications.

   This Specification is not applicable in OPERATIONAL MODE 4 or 5.

D. Entry into an OPERATIONAL MODE or other specified condition shall not be made when the conditions for the Limiting Conditions for Operation are not met and the associated ACTION requires placing the plant in an OPERATIONAL MODE or other specified condition of operation in which the Limiting Condition for Operation does not apply if they are not met within a specified time interval. Entry into an OPERATIONAL MODE or other specified condition may be made in accordance with the ACTION requirements when conformance to them permits continued operation of the facility for an unlimited period of time. This provision shall not prevent passage through or to OPERATIONAL MODE(s) as required to comply with ACTION requirements. Exceptions to these requirements are stated in the individual Specifications.

4.0 - SURVEILLANCE REQUIREMENTS

A. Surveillance Requirements shall be met during the reactor OPERATIONAL MODE(s) or other conditions specified for individual Limiting Conditions for Operation unless otherwise stated in an individual Surveillance Requirement.

B. Each Surveillance Requirement shall be performed within the specified surveillance interval with a maximum allowable extension not to exceed 25 percent of the surveillance interval.

C. Failure to perform a Surveillance Requirement within the allowed surveillance interval, defined by Specification 4.0.B, shall constitute noncompliance with the OPERABILITY requirements for a Limiting Condition for Operation. The time limits of the ACTION requirements are applicable at the time it is identified that a Surveillance Requirement has not been performed. The ACTION requirements may be delayed for up to 24 hours to permit the completion of the surveillance when the allowable outage time limits of the ACTION requirements are less than 24 hours. Surveillance requirements do not have to be performed on inoperable equipment.

D. Entry into an OPERATIONAL MODE or other specified applicable condition shall not be made unless the Surveillance Requirement(s) associated with the Limiting Condition for Operation have been performed within the applicable surveillance interval or as otherwise specified. This provision shall not prevent passage through or to OPERATIONAL MODE(s) as required to comply with ACTION requirements.

E. Surveillance Requirements for inservice inspection and testing of ASME Code Class 1, 2, and 3 components shall be applicable as follows:

1. Inservice Inspection of ASME Code Class 1, 2, and 3 components and inservice testing of ASME Code Class 1, 2, and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR Part 50, Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 50, Section 50.55a(g)(6)(i).

2. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice inspection and testing activities required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these Technical Specifications:

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice inspection and testing activities	Required Frequencies for performing inservice inspection and testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days

3. The provisions of Specification 4.0.B are applicable to the above required frequencies for performing inservice inspection and testing activities.

4. Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements.

5. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any Technical Specification.

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

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Specifications 3.0.A through 3.0.D establish the general requirements applicable to Limiting Conditions for Operation. These requirements are based on the requirements for Limiting Conditions for Operation stated in the Code of Federal Regulations, 10 CFR 50.36(c)(2):

> "Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specification until the condition can be met."

Specification 3.0.A establishes the Applicability statement within each individual specification as the requirement for when (i.e., in which OPERATIONAL MODE(s) or other specified conditions) conformance to the Limiting Conditions for Operation is required for safe operation of the facility. The ACTION requirements establish those remedial measures that must be taken within specified time limits when the requirements of a Limiting Condition for Operation are not met. It is not intended that the shutdown ACTION requirements be used as an operational convenience which permits (routine) voluntary removal of a system(s) or component(s) from service in lieu of other alternatives that would not result in redundant systems or components being inoperable.

There are two basic types of ACTION requirements. The first specifies the remedial measures that permit continued operation of the facility which is not further restricted by the time limits of the ACTION requirements. In this case, conformance to the ACTION requirements provides an acceptable level of safety for unlimited continued operation as long as the ACTION requirements continue to be met. The second type of ACTION requirement specifies a time limit in which conformance to the conditions of the Limiting Condition for Operation must be met. This time limit is the allowable outage time to restore an inoperable system or component to OPERABLE status or for restoring parameters within specified limits. If these ACTION(s) are not completed within the allowable outage time limits, a shutdown is required to place the facility in a reactor OPERATIONAL MODE or other specified condition in which the specification no longer applies.

The specified time limits of the ACTION requirements are applicable from the point in time it is identified that a Limiting Condition for Operation is not met. The time limits of the ACTION requirements are also applicable when a system or component is removed from service for surveillance testing or investigation of operational problems. Individual specifications may include a specified time limit for the completion of a Surveillance Requirement when equipment is removed from service. In this case, the allowable outage time limits of the ACTION requirements are applicable when this limit expires if the surveillance has not been completed. When a shutdown is required to comply with ACTION requirements, the plant may have entered an OPERATIONAL MODE in which a new specification becomes applicable. In this case, the time limits of the ACTION requirements would apply from the point in time that the new specification becomes applicable if the requirements of the Limiting Condition for Operation are not met.

Specification 3.0.B establishes that noncompliance with a specification exists when the requirements of the Limiting Condition for Operation are not met and the associated ACTION

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requirements have not been implemented within the specified time interval. The purpose of this specification is to clarify that (1) implementation of the ACTION requirement within the specified time interval constitutes compliance with a specification and (2) completion of the remedial measures of the ACTION requirements is not required when compliance with a Limiting Condition for Operation is restored within the time interval specified in the associated ACTION requirements.

Specification 3.0.C establishes the shutdown ACTION requirements that must be implemented when a Limiting Condition for Operation is not met and the condition is not specifically addressed by the associated ACTION requirements. The purpose of this specification is to delineate the time limits for placing the unit in a safe shutdown condition when plant operation cannot be maintained within the limits for safe operation defined by the Limiting Condition for Operation and its ACTION requirements. It is not intended to be used as an operational convenience which permits (routine) voluntary removal of redundant systems or components from service in lieu of other alternatives that would not result in redundant systems or components being inoperable. One hour is allowed to prepare for an orderly shutdown before initiating a change in plant operation. This time permits the operator to coordinate the reduction in electrical generation with the load dispatcher to ensure the stability and availability of the electrical grid. The time limits specified to reach lower MODE(s) of operation permit the shutdown to proceed in a controlled and orderly manner that is well within the specified maximum cooldown rate and within the cooldown capabilities of the facility assuming only the minimum required equipment is OPERABLE. This reduces thermal stresses on components of the primary coolant system and the potential for a plant transient that could challenge safety systems under conditions for which this specification applies.

If remedial measures permitting limited continued operation of the facility under the provisions of the ACTION requirements are completed, the shutdown may be terminated. The time limits of the ACTION requirements are applicable from the point in time there was a failure to meet a Limiting Condition for Operation. Therefore, the shutdown may be terminated if the ACTION requirements have been met or the time limits of the ACTION requirements have not expired, thus providing an allowance for the completion of the required ACTION(s).

The time limits of Specification 3.0.C allow 37 hours for the plant to be in COLD SHUTDOWN when a shutdown is required during POWER OPERATION. If the plant is in a lower MODE of operation when a shutdown is required, the time limit for reaching the next lower MODE applies. However, if a lower MODE of operation is reached in less time than allowed, the total allowable time to reach COLD SHUTDOWN, or other OPERATIONAL MODE, is not reduced. For example, if HOT SHUTDOWN is reached in 10 hours, the time allowed to reach COLD SHUTDOWN is the next 27 hours because the total time to reach COLD SHUTDOWN is not reduced from the allowable limit of 37 hours. Therefore, if remedial measures are completed that would permit a return to POWER OPERATION, a penalty is not incurred by having to reach a lower MODE of operation in less than the total time allowed.

The same principle applies with regard to the allowable outage time limits of the ACTION requirements, if compliance with the ACTION requirements for one specification results in entry into an OPERATIONAL MODE or condition of operation for another specification in which the

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requirements of the Limiting Condition for Operation are not met. If the new specification becomes applicable in less time than specified, the difference may be added to the allowable outage time limits of the second specification. However, the allowable outage time limits of ACTION requirements for a higher MODE of operation may not be used to extend the allowable outage time that is applicable when a Limiting Condition for Operation is not met in a lower MODE of operation.

The shutdown requirements of Specification 3.0.C do not apply in MODES 4 or 5, because the ACTION requirements of individual specifications define the remedial measures to be taken.

Specification 3.0.D establishes limitations on a change in OPERATIONAL MODE(s) when a Limiting Condition for Operation is not met. It precludes placing the facility in a higher MODE of operation when the requirements for a Limiting Condition for Operation are not met and continued noncompliance to these conditions would result in placing the plant in an OPERATIONAL MODE or other specified condition of operation in which the Limiting Condition for Operation does not apply to comply with the ACTION requirements if a change in MODE(s) were permitted. The purpose of this specification is to ensure that facility operation is not initiated or that higher MODE(s) of operation or other specified conditions are not entered when corrective ACTION is being taken to obtain compliance with a specification by restoring equipment to OPERABLE status or parameters to specified limits. Compliance with ACTION requirements that permit continued operation of the facility for an unlimited period of time provides an acceptable level of safety for continued operation without regard to the status of the plant before or after a change in OPERATIONAL MODE(s). Therefore, in this case, entry into an OPERATIONAL MODE or other specified condition may be made in accordance with the provisions of the ACTION requirements. The provisions of this specification should not, however, be interpreted as endorsing the failure to exercise good practice in restoring systems or components to OPERABLE status before plant startup.

When a shutdown is required to comply with ACTION requirements, the provisions of Specification 3.0.D do not apply because they would delay placing the facility in a lower MODE of operation.

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Specifications 4.0.A through 4.0.E establish the general requirements applicable to Surveillance Requirements. These requirements are based on the Surveillance Requirements stated in the Code of Federal Regulations, 10 CFR 50.36(c)(3):

> "Surveillance requirements are requirements relating to test, calibration, or inspection to ensure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions of operation will be met."

<u>Specification 4.0.A</u> establishes the requirement that surveillances must be performed during the OPERATIONAL MODE(s) or other conditions for which the requirements of the Limiting Condition for Operation apply unless otherwise stated in an individual Surveillance Requirement. The purpose of this specification is to ensure that surveillances are performed to verify the operational status of systems and components and that parameters are within specified limits to ensure safe operation of the facility when the plant is in a reactor OPERATIONAL MODE or other specified condition for which the individual Limiting Condition for Operations are applicable. Surveillance Requirements do not have to be performed when the facility is in an OPERATIONAL MODE for which the requirements of the associated Limiting Condition for Operation do not apply unless otherwise specified. The Surveillance Requirements associated with a Special Test Exception are only applicable when the Special Test Exception is used as an allowable exception to the requirements of a specification.

<u>Specification 4.0.B</u> establishes the limit for which the specified time interval for Surveillance Requirements may be extended. It permits an allowable extension of the normal surveillance interval to facilitate surveillance scheduling and consideration of plant operating conditions that may not be suitable for conducting the surveillance; e.g., transient conditions or other ongoing surveillance or maintenance activities. It also provides flexibility to accommodate the length of a fuel cycle for surveillances that are specified with and 18 month surveillance interval. It is not intended that this provision be used repeatedly as a convenience to extend surveillance intervals beyond that specified for surveillances that are not performed during refueling outages. Likewise, it is not the intent that refueling outage surveillances be performed during power operation unless it is consistent with safe plant operation. The limitation of Specification 4.0.B is based on engineering judgment and the recognition that the most probable result of any particular surveillance being performed is the verification of conformance with the Surveillance Requirements. This provision is sufficient to ensure that the reliability ensured through surveillance activities is not significantly degraded beyond that obtained from the specified surveillance interval.

<u>Specification 4.0.C</u> establishes that the failure to satisfy a Surveillance Requirement within the allowed surveillance interval, defined by the provisions of Specification 4.0.B, is a condition that constitutes a failure to meet the OPERABILITY requirements for a Limiting Condition for Operation. Under the provisions of this specification, systems and components are assumed to be OPERABLE when Surveillance Requirements have been satisfactorily performed within the specified time

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interval. However, nothing in this provision is to be construed as implying that systems or components are OPERABLE when they are found or known to be inoperable although still meeting the Surveillance Requirements. This specification also clarifies that the ACTION requirements are applicable when Surveillance Requirements have not been completed within the allowed surveillance interval and that the time limits of the ACTION requirements apply from the point in time it is identified that a surveillance has not been performed and not at the time that the allowed surveillance interval was exceeded. Completion of the Surveillance Requirement within the allowable outage time limits of the ACTION requirements restores compliance with the requirements of Specification 4.0.C. However, this does not negate the fact that the failure to have performed the surveillance within the allowed surveillance interval, defined by the provisions of Specification 4.0.B, was a violation of the OPERABILITY requirements for a Limiting Condition for Operation that is subject to enforcement action. The failure to perform a surveillance within the provisions of Specification 4.0.B is a violation of a Technical Specification requirement and is, therefore, a reportable event under the requirements of 10 CFR 50.73(a)(2)(i)(B) because it is a condition prohibited by the plant's Technical Specifications.

If the allowable outage time limits of the ACTION requirements are less than 24 hours or a shutdown is required to comply with ACTION requirements (e.g., in Specification 3.0.C), a 24-hour allowance is provided to permit a delay in implementing the ACTION requirements. This provides an adequate time limit to complete Surveillance Requirements that have not been performed. The purpose of this allowance is to permit the completion of a surveillance before a shutdown would be required to comply with ACTION requirements or before other remedial measures would be required that may preclude the completion of a surveillance. The basis for this allowance includes consideration for plant conditions, adequate planning, availability of personnel, the time required to perform the surveillance, and the safety significance of the delay in completing the required surveillance. This provision also provides a time limit for the completion of Surveillance Requirements that become applicable as a consequence of MODE changes imposed by ACTION requirements and for completing Surveillance Requirements that are applicable when an exception to the requirements of Specification 4.0.D is allowed. If a surveillance is not completed within the 24-hour allowance, the time limits of the ACTION requirements are applicable at that time. When a surveillance is performed within the 24-hour allowance and the Surveillance Requirements are not met, the time limits of the ACTION requirements are applicable at the time that the surveillance is terminated.

Surveillance Requirements do not have to be performed on inoperable equipment because the ACTION requirements define the remedial measures that apply. However, the Surveillance Requirements have to be met to demonstrate that inoperable equipment has been restored to OPERABLE status.

<u>Specification 4.0.D</u> establishes the requirement that all applicable surveillances must be met before entry into an OPERATIONAL MODE or other condition of operation specified in the Applicability statement. The purpose of this specification is to ensure that system and component OPERABILITY requirements or parameter limits are met before entry into an OPERATIONAL MODE or other specified condition for which these systems and components ensure safe operation of the

facility. This provision applies to changes in OPERATIONAL MODE(s) or other specified conditions associated with plant shutdown as well as startup.

Under the provisions of this specification, the applicable Surveillance Requirements must be performed within the specified surveillance interval to assure that the Limiting Conditions for Operation are met during initial plant startup or following a plant outage.

When a shutdown is required to comply with ACTION statements, the provisions of Specification 4.0.D do not apply because this would delay placing the facility in a lower MODE of operation.

Specification 4.0.E establishes the requirement that inservice inspection of ASME Code Class 1, 2, and 3 components and inservice testing of ASME Code Class 1, 2, and 3 pumps and valves shall be performed in accordance with a periodically updated version of Section XI of the ASME Boiler and Pressure Vessel Code and Addenda as required by 10 CFR 50.55a. These requirements apply except when relief has been provided in writing by the Commission.

This specification includes a clarification of the frequencies for performing the inservice inspection and testing activities required by Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda. This clarification is provided to ensure consistency in surveillance intervals throughout the Technical Specifications and to remove any ambiguities relative to the frequencies for performing the required inservice inspection and testing activities.

Under the terms of this specification, the more restrictive requirements of the Technical Specifications take precedence over the ASME Boiler and Pressure Vessel Code and applicable Addenda. The requirements of Specification 4.0.D to perform surveillance requirements before entry into an OPERATIONAL MODE or other specified condition takes precedence over the ASME Boiler and Pressure Vessel Code provision that allows pumps and valves to be tested up to one week after return to normal operation. The Technical Specification definition of OPERABLE does not allow a grace period before a component, which is not capable of performing its specified function, is declared inoperable and takes precedence over the ASME Boiler and Pressure Vessel Code provision that allows a valve to be incapable of performing its specified function for up to 24 hours before being declared inoperable.

ATTACHMENT 4

EXISTING TECHNICAL SPECIFICATIONS

Technical Specification 3/4.0

"APPLICABILITY"

ATTACHMENT 4

DELETION OF CURRENT TECHNICAL SPECIFICATIONS

This technical specification amendment will replace the current section 3/4.0, Applicability, for Dresden Unit 2 and Unit 3 Technical Specifications. The specifications are replaced in there entirety with revised pages that combine the Unit 2 and Unit 3 specifications.

Delete the following pages:

DPR - 19	DPR - 25
3.0-1	3.0-1
3.0-2	3.0-2
B3.0-3	B3.0-3

ATTACHMENT 4

DELETION OF CURRENT TECHNICAL SPECIFICATIONS

This technical specification amendment will replace the current section 3/4.0, Applicability, for Quad Cities Unit 1 and Unit 2 Technical Specifications. The specifications are replaced in there entirety with revised pages that combine the Unit 1 and Unit 2 specifications.

Delete the following pages:

DPR - 29	DPR - 30
3.0/4.0-1	3.0/4.0-1
3.0/4.0-2	3.0/4.0-2
	3.0/4.0-3

ATTACHMENT 5

DRESDEN 2/3 DIFFERENCES

Technical Specification 3/4.0

"APPLICABILITY"

ATTACHMENT 5

COMPARISON OF DRESDEN UNIT 2 AND UNIT 3 TECHNICAL SPECIFICATIONS
FOR THE
IDENTIFICATION OF TECHNICAL DIFFERENCES

SECTION 3.0/4.0
"APPLICABILITY"

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.) or punctuation but rather to identify areas which the Technical Specifications are technically different.

The review of the Section 3.0/4.0, "Applicability" did not identify any technical differences.

ATTACHMENT 5

QUAD CITIES 1/2 DIFFERENCES

Technical Specification 3/4.0

"APPLICABILITY"

ATTACHMENT 5

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

SECTION 3.0/4.0
"APPLICABILITY"

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.) or punctuation but rather to identify areas which the Technical Specifications are technically different.

The review of the Section 3.0/4.0, "Applicability" did not identify any technical differences.

ATTACHMENT 6

SIGNIFICANT HAZARDS CONSIDERATIONS AND ENVIRONMENTAL ASSESSMENT EVALUATION

Technical Specification 3/4.0

"APPLICABILITY"

ATTACHMENT 6

EVALUATION OF SIGNIFICANT HAZARDS CONSIDERATION

Commonwealth Edison has evaluated this proposed amendment and determined that it involves no significant hazards considerations. According to 10 CFR 50.92(c), a proposed amendment to an operating license involves no significant hazards considerations 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:

The changes in this proposed amendment are more restrictive than present Technical Specification requirements. These more restrictive requirements will help to ensure that the intent of plant operating philosophy embodied in the Standard Technical Specifications is included in the Dresden and Quad Cities Technical Specifications. the inclusion of these requirements will also provide clarification to the plant operating staff and help to prevent misinterpretations. The changes are modeled after those of the present BWR STS as modified by Generic Letters (primarily Generic Letter 87-09). Since this amendment request retains the provisions of present LCO 3.0, adds more restrictive provisions, and provides administrative changes to correct inconsistencies found in the STS, there is no significant increase in the probability or consequences of an accident previously evaluated.

This proposed amendment request includes the provisions of Generic Letter 89-14 which deletes the 3.25 limit from specification 4.0.B. This deletion has been reviewed by the NRC and found to result in a greater benefit to safety. The NRC concluded that the provisions of the letter will remove an unnecessary restriction of extending surveillance requirements and will result in a benefit to safety when plant conditions are not conducive to the safe conduct of surveillance requirements. The NRC further concluded that removal of the 3.25 limit will provide greater flexibility in the use of the provision for extending surveillance intervals, reduce the administrative burden associated with its use, and have a positive effect on safety. Commonwealth Edison agrees with the NRC conclusions and also concludes that the provisions of Generic Letter 89-14 should be implemented at Dresden and Quad Cities Stations.

This proposed amendment request also includes provisions from Generic Letters 88-01 and a minor clarification of the wording proposed by Generic Letter 87-09. These provisions provide for inclusion of additional inspection requirements in the inservice testing program for intergranular stress corrosion cracking of piping, and for inclusion of a clarification of application of Technical Specification 3.0.D based on the text of the Generic Letter. These changes also provide additional restrictions on the operation of the plant to improve safety as discussed in each of the Generic Letters. The inclusion of the changes from Generic Letters 87-09, 88-01, 89-14, and 91-04 does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated because:

The proposed changes are a complete adoption of the STS and Generic Letters 87-09 and 89-14 LCOs and surveillance requirements. The changes embody present operating philosophy contained in the individual specifications and add Generic Letters 87-09, 88-01 and 89-14 provision which have been evaluated by the NRC and found acceptable for inclusion in the Technical Specifications. The proposed changes do not allow any new modes of plant operation and therefore, no changes are involved that would create a new or different type of accident that previously evaluated.

The proposed changes do not involve a significant reduction in a margin of safety because:

The more restrictive provisions proposed to be added by this amendment will increase the margin of safety by clearly defining to the plant operating personnel the governing LCO and surveillance requirement provisions. These new provisions will help to prevent misinterpretation where no requirements are presently stated. Since more restrictive requirements are proposed, this amendment does not involve a significant reduction in the margin of safety. The provisions of Generic Letters 87-09, 88-01 and 89-14 have been evaluated by the NRC and found acceptable for plant use. In addition, Commonwealth Edison has evaluated these provisions for inclusion at Dresden and Quad Cities Stations and have concluded that the margin of safety is preserved or improved by using the proposed changes.

Conclusion

Guidance has been provided in "Final Procedures and Standard on No Significant Hazards Considerations," Final Rule 51 FR 7744, for the application of standards to license change requests for determination of the existence of significant hazards considerations. This document provides examples of amendments which are and are not considered likely to involve significant hazards considerations. These proposed amendments most closely fit examples (i), (ii), and (Vii) of amendments that are considered not likely to involve significant hazards considerations.

The proposed amendments do not involve a significant relaxation of the criteria used to establish safety limits, a significant relaxation of the bases for the limiting safety system settings or a significant relaxation of the bases for the limiting conditions for operations. Therefore, based on the guidance provided in the Federal Register and the criteria established in 10 CFR 50.92(c), the proposed change does not constitute a significant hazards consideration.

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 1

EXECUTIVE SUMMARY

Technical Specification 3/4.3

"REACTIVITY CONTROL"

EXECUTIVE SUMMARY

The Dresden Technical Specification Upgrade Program (TSUP) was conceptualized in response to lessons learned from the Diagnostic Evaluation Team inspection and the frequent need for Technical Specification interpretations. A comparison study of the Standard Technical Specification (STS), later operating plant's Technical Specifications provisions and Quad Cities Technical Specifications was performed prior to the Dresden TSUP effort. The study identified potential improvements in clarifying requirements and requirements which are no longer consistent with current industry practices. The Dresden TSUP will enhance the Quad Cities TSUP currently under review by the NRC. As a result of the inconsistencies in the Quad Cities submittal compared to the Standard Technical Specifications (STS), Dresden's submittal will more closely follow the provisions of STS and in conjunction, Quad Cities will amend their submittals so that Quad Cities and Dresden are identical within equipment and plant design. The format for the Dresden TSUP will remain as a two column layout for human factors considerations. Additionally, chapter organizations will remain unchanged.

The TSUP is not intended to be a complete adoption for the STS. Overall, the Dresden custom Technical Specifications provide for the safe operation of the plant and therefore, only an upgrade is deemed necessary.

In response to an NRC recommendation, Quad Cities combined the Unit 1 and Unit 2 Technical Specifications into one document. The Dresden Unit 2 and Unit 3 Technical Specifications will also be combined into one document. To accomplish the combination of the Units' Technical Specification, a comparison of the Unit 2 and Unit 3 Technical Specification was performed to identify any technical differences. The technical differences are identified in the proposed amendment package for each section.

The TSUP was identified as a station top priority and is currently contained in the Dresden Management Action Plan (DMAP). The TSUP goal is to provide a better tool to station personnel to implement their responsibilities and to ensure Dresden Station is operated in accordance with current industry practices. The improved Technical Specifications provide for enhanced operation of the plant. The program improves the operator's ability to use the Technical Specifications by more clearly defining the Limiting Conditions for Operation and required actions. The most significant improvement to the specifications is the addition of equipment operability requirements during shutdown conditions.

EXECUTIVE SUMMARY
(continued)
PROPOSED CHANGES TO TECHNICAL SPECIFICATION SECTION 3/4.3, "REACTIVITY CONTROL"

The proposed changes delete the present Objective statements and provides Applicability statements within each specification in accordance with STS guidelines. The proposed Applicability statements include the Operation Modes or other conditions for which the LCO must be satisfied. An STS type of format is proposed which retaining the present two column layout.

The specification in Section 3/4.3 have been reordered and new titles have been added based on STS arrangements and nomenclature.

Present Specification 3.3.A, Reactivity Limitations, is retitled as Shutdown Margin in accordance with STS guidelines. Action statements are proposed based on STS guidelines. Additional surveillances for Section 4.3.A implement STS and later operating BWR plant provisions for checking Shutdown Margin when withdrawn control rods are inoperable due to being immovable.

Proposed Specification 3/4.3.B, Reactivity Anomalies, replaces present specifications. Tr proposed Action statements follow STS guidelines.

The proposed LCO for Specification 3.3.C is taken from STS guidelines. The present provision which allows control rods that are fully inserted and disarmed to not be considered inoperable, is deleted. The proposed Applicability and Action statements implement STS guidelines and replace present requirements.

In accordance with STS guidelines, three new sections: 3/4.3.D, Control Rod Maximum Scram Insertion Times; 3/4.3.E, Control Rod Average Scram Insertion Times; and, 3/4.3.F, Four Control Rod Group Scram Insertion Times, are proposed as replacements for present Specifications on Scram Insertion Times.

Proposed Specification 3/4.3.G, Control Rod Scram Accumulators, is a rewrite of present Specifications. The proposed Actions for inoperable control rod scram accumulators implement STS guidelines. The proposed Surveillance Requirements incorporate the weekly STS testing guidelines.

Proposed Specification 3/4.3.H, Control Rod Drive Coupling, is a rewrite of existing Specifications. Proposed Action steps for uncoupled rods follow STS guidelines. Present Surveillance Requirements are changed to match STS guidelines and format.

Proposed Specification 3/4.3.I, Control Rod Position Indication System, is a rewrite of existing specifications for Quad Cities and is a new section for Dresden Station. The Actions a d Surveillance Requirements are based on STS guidelines and provisions.

EXECUTIVE SUMMARY
(continued)
PROPOSED CHANGES TO TECHNICAL SPECIFICATION
SECTION 3/4.3, "REACTIVITY CONTROL"

Proposed Specification 3/4.3.J, Control Rod Drive Housing Supports, is a rewrite of existing Specifications. The proposed LCO and Applicability follow STS guidelines and implement present Technical Specification provisions. The proposed Action follows STS guidelines. The present surveillance requirement is rewritten using STS wording.

Proposed Specification 3/4.3.K, Scram Discharge Volume Vent and Drain Valves, is a rewrite of present Specifications which address only the Surveillance Requirements. The present Specifications do not contain LCO, Applicability or Action provisions. The rewrite uses STS provisions to address the missing requirements.

Proposed Specification 3/4.3.L, Rod Worth Minimizer, is a rewrite of present Specifications. The proposed LCO, Applicability, Actions, and Surveillance Requirements are taken from STS guidelines.

Proposed Specification 3/4.3.M, Rod Block Monitor, is a rewrite of present Specifications. The proposed LCO implements requirements and Action steps based on STS guidelines. The proposed Surveillance Requirements are taken from present Technical Specification requirements which coincide with STS guidelines.

Proposed Specification 3/4.3.N on the Economic Generation Control System retains the present provisions. STS format and wording style are used for consistency with other proposed changes.

The proposed changes to the 3/4.3 Bases are made to support the changes proposed to the individual specifications.

ATTACHMENT 2

DESCRIPTION OF CHANGES

Technical Specification 3/4.3

"REACTIVITY CONTROL"

ATTACHMENT 2

System retains the present provisions for EGC operation. STS format and wording style are used for consistency with other proposed changes.

The proposed changes to the Limiting Conditions for Operation Bases 3.3 are made to support the changes proposed to the individual specifications. The proposed Bases are a combination of the STS Bases, the current Dresden Bases, the current Quad Cities Bases and include addtional information as determined necessary.

ATTACHMENT 2

DESCRIPTION OF PROPOSED AMENDMENT

The changes proposed in this amendment request are made to improve the understanding and usability of the present technical specifications, and to incorporate the technical improvements from the Standard Technical Specifications (STS).

The present Dresden and Quad Cities Technical Specifications contain Applicability and Objective statements at the beginning of most sections. These statements are generic in nature and do not provide any useful information to the user of the technical specifications. The proposed change will delete the Objective statement and provide Applicability statements within each specification based on the STS. The proposed Applicability statement to be included in each specification will include the applicable operational modes or other conditions for which the Limiting Condition for Operation (LCO) must be satisfied.

The STS action provisions which delineate a specification 3.0.4 exception are not incorporated into the proposed specifications. The incorporation of the Generic Letter 87-09 change to STS specification 3.0.4 (Dresden and Quad Cities proposed 3.0.D specification) requires that each action be independently evaluated for applicability of the new specification. These evaluations are provided in Attachment 7.

<u>Section 3/4.3.A, Shutdown Margin</u>

This section is a complete adoption of the STS with two exceptions. The proposed changes include adding appropriate action, Applicability, and Surveillance Requirements (SR) from STS guidelines. Present Specification, Reactivity Limitations, is re-titled as shutdown margin in accordance with STS guidelines. The proposed LCO for Specification 3.3.A is based on STS requirements with the shutdown margin limitation of 0.25% delta k/k currently stated in the TS. A new LCO is added in accordance with the STS. The new LCO requires demonstration of shutdown margin by analytical methods. The amount of shutdown margin demonstrated analytically is greater than that demonstrated by test. This proposed change to the LCO appropriately places the limiting condition in the LCO instead of the SR. The present applicability for the shutdown margin technical specifications is all operational modes. The proposed change implements this requirement by requiring operability in operational modes 1, 2, 3, 4 and 5. The present specification has no action provisions and thus, action statements are proposed based on STS guidelines. Action 3.3.A.1 addresses operational modes 1 or 2 and requires restoration of the required shutdown margin within 6 hours or be in at least hot shutdown within the next 12 hours. Action 3.3.A.2 addresses operational modes 3 or 4 and requires immediate verification that all insertable control rods are inserted and suspension of all activities that could reduce the shutdown margin. Proposed action 3.3.A.2 also requires establishment of secondary containment integrity within 8 hours. Proposed action 3.3.A.3 addresses operational mode 5 and requires suspension of core alterations and oth activities that could reduce the shutdown margin and insertion of all insertable control rods within 1 hour. Proposed action 3.3.A.3 also requires establishment of secondary containment integrity within 8 hours. The proposed action provisions address all applicable operational modes and ensure, through limited time for restoration, that shutdown margin limitations are enforced or proper remedial measures are followed. The proposed SRs implement the present requirement to demonstrate shutdown margin provisions during the first startup following a refueling outage in which core alterations were performed. One exception to the STS is that the proposed changes do not include STS SR 4.3.A.1.b, which requires the demonstration of shutdown margin within 500 MWD/T of the point in the cycle where the minimum shutdown margin is equal to the shutdown margin limit. Advances in fuel designs now allow reactor cores to be loaded with upwards of 3% shutdown margin. This SR would only represent an additional administrative burden and adds no value to the shutdown margin requirement. Therefore, this SR is expected to never be required. Added to the present SRs is 4.3.A.1.b, which provides the demonstration of shutdown margin within 24 hours after detection of a withdrawn control rod that is immovable, as a result of excessive friction or mechanical interference, or is known to be untrippable, except that the shutdown margin shall be verified acceptable with the increased worth of the inoperable control rod. Current STS provisions require the shutdown margin demonstration to be completed within 12 hours of the detection of an immovable control rod. 24 hours is proposed as a result of the minimum required time to perform the shutdown margin calculations and have them approved inaccordance with station procedures. A new SR is added to verify analytically that the shutdown margin has been demonstrated analytically prior to performing core alterations. This new SR will provide for shutdown margin analytical determination until such time that plant conditions allow for the shutdown margin demonstration by test.

Section 3/4.3.B, Reactivity Anomalies

This section is a complete adoption of the STS. Proposed specification 3/4.3.B on Reactivity Anomalies replaces present Specifications for Reactivity Anomalies. The proposed LCO is taken from STS provisions and requires the rod density of the difference between the actual critical rod density and the predicted rod density shall not exceed 1% delta k/k. Dresden compares the critical rod configuration with the predicted rod configuration. This difference is a result of the fuel vendors preferences. The present Applicability of during power operation is implemented by requiring operability in operational modes 1 and 2. The present requirement to notify the NRC within 24 hours in accordance with Specification 6.6 if the 1% limit is exceeded, is beyond the standard reporting requirements. Since there are no plant specific differences to provide a bases for this report it is being deleted. The proposed action statements follow STS guidelines. The proposed actions allow 12 hours to perform an analysis to determine and explain the cause of the reactivity difference and then operation may continue if the difference is explained and corrected. If these conditions are not met, then the reactor must be in at least hot shutdown within the next 12 hours. The proposed SR 4.3.B requires reactivity anomaly tests during the first startup following core alterations and at least once per 31 effective full power days.

Section 3/4.3.C, Control Rod operability

This section is a complete adoption of the STS with one enhancement. This item details the rewrite of present Specification for Inoperable Control Rods, and its relocation to proposed Specification 3/4.3.C, Control Rod operability. Reordering of the specifications and the new titles are based on STS arrangements and nomenclature. The proposed LCO for Specification 3.3.C, Control Rod operability, is taken from STS guidelines. All control rods are required to be operable in the applicable operational modes as stated in the STS. The present provision which allows control rods that are fully inserted and disarmed to not be considered inoperable, is deleted. The proposed Applicability for Specification 3.3.C, of operational modes 1 and 2, is taken from STS guidelines. Present specifications indicate that operability is required during reactor power operation and thus the use of operational modes 1 and 2 meets the intent of these present specifications. The proposed actions for Control Rod operability implement STS guidelines and replace present requirements. Proposed action 3.3.C.1 addresses the condition where one control rod is inoperable due to being immovable, as a result of excessive friction or mechanical interference, or known to be untrippable. Proposed action 3.3.C.1 requires within one hour, verification that the inoperable rod, if withdrawn, is separated from all other inoperable withdrawn control rods by at least two control cells in all directions, that the control rod be disarmed electrically or hydraulically, and that compliance with SR 4.3.A.2 be initiated. If these conditions are not met, the reactor is required to be in at least hot shutdown within the next 12 hours. STS require that an inoperable control rod be restored to operable status if it is withdrawn within 48 hours or the reactor is required to be in at least hot shutdown within the next 12 hours. The requirement was not included in the proposed specifications because it represents an ambiguous step. If the SRs are satisfied, there is no technical reason why the reactor needs to be shutdown. The addition of specifying that the rod is withdrawn is made to prevent an unnecessary reactor shutdown if a control rod is stuck at the full-in position. Proposed action 3.3.C.2 addresses the condition with one or more control rods trippable but inoperable for causes other than that addressed in action 3.3.C.1. Proposed action 3.3.C.2 requires that for withdrawn control rods, within 1 hour, that inoperable control rods be verified to be separated from all other inoperable withdrawn control rods by at least two control cells in all directions and demonstration of the insertion capability of the inoperable withdrawn control by inserting the control rods at least one notch. Proposed action 3.3.C.2 further requires that with the provisions of this action not met, the inoperable withdrawn control rods be inserted and disarmed. Provisions are added from the STS to allow the disarmed directional control valves to be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rods to operable status. The present reference to a failed control rod drive collet housing is outdated and is being deleted from the Dresden and Quad Cities Technical Specifications. Collet housing cracking and potential failures are detected during control rod disassembly and inspection conducted during regular refueling outages. Proposed Surveillance Requirements 4.3.C adopt STS provisions to exercise control rods weekly and daily. Implementing STS SR provisions ensures that daily control rod notch movement tests are performed only when there is a demonstrated inability of a control rod to be inserted in the reactor core. Proposed SR 4.3.C.2 is based on STS guidelines and helps to ensure understanding of control rod operability by referencing performance of SRs 4.3.D, 4.3.G, 4.3.H, and 4.3.I.

The next sections provide descriptions of the separation of the present Specification on Scram Insertion Times into three separate specifications in accordance with STS guidelines. Proposed Specification 3/4.3.D on Control Rod Maximum Scram Insertion Times, proposed Specification 3/4.3.E on Control Rod Average Scram Insertion Times, and proposed Specification 3/4.3.F on Four Control Rod Group Scram Insertion Times are adopted from the STS.

Section 3/4.3.D, Control Rod Maximum Scram Insertion Times

This section is a complete adoption of the STS with one clarification. Proposed Specification 3/4.3.D on Control Rod Maximum Scram Insertion Times is written using STS provisions which require that the maximum scram insertion time for 90% insertion of any operable control rod shall not exceed 7 seconds. The proposed Applicability of operational modes 1 and 2 implements STS guidelines and requires maximum scram insertion times to be met in the reactor conditions where multiple rods can be withdrawn and where accident analyses depend on these insertion limits. Proposed action steps for Specification 3.3.D are based on STS. Present Specifications require control rods not meeting the maximum scram insertion time limits to be declared inoperable, fully inserted into the core and electrically disarmed. Proposed action 3.3.D.1 require the affected control rod to be declared inoperable and require performance of SR 4.3.D.2 at least once per 60 days when operation is continued with three or more control rods with maximum scram insertion times exceeding 7 seconds. The STS action for control rods that do not meet the scram insertion time limit was clarified to state more clearly what a slow control rod entails. SR 4.3.D.3

requires maximum scram time tests for at least 10% of the control rods. If action 3.3.D.1 and 2 are not met then the reactor is required to be in at least hot shutdown within the next 12 hours. The Surveillance Requirements proposed for the Control Rod Maximum Scram Insertion Time, 4.3.D, incorporate requirements based on STS guidelines. Present provisions to perform scram time tests on all control rods after a refueling and before exceeding 30% power are modified in proposed SR 4.3.D.1 by requiring the SR prior to exceeding 40% power following core alterations or after a reactor shutdown that is greater than 120 days. The power level is raised in accordance with the STS and justified on the fact that the minimum critical power ratio is extremely non-limiting at the 40% power level. The higher power level will afford Dresden and Quad Cities additional margin from the Banked Position Withdrawal Sequence low power setpoint to perform the scram timing. Scram time testing is added by SR 4.3.D.2 for individual control rods following maintenance or modification work that could affect scram times. These changes will ensure testing in situations that can directly affect control rod insertion times. A footnote is added to the proposed SR to allow operational mode changes prior to performing the required SR for individual control rods that have had maintenance performed. Current specifications require that all control rods be scram time tested after each refueling outage and that 50% of the control rods be measured for scram times not more frequently than 16 weeks nor less frequently than 32 weeks. These present requirements are replaced with proposed SR 4.3.D.3 which is STS based and requires at least 10% of the control rods, on a rotating basis, to be scram time tested at least once per 120 days of reactor power operation. The scram time testing of proposed SR 4.3.D.3 has been proven successful through use for detecting scram time deterioration at operating BWRs with control rod drive systems similar in design to that of Dresden and Quad Cities. The population of the control rods subjected to scram timing will be reduced as a result of adopting the STS SR for scram timing. The reduction does not have an effect on the Minimum Critical Power Ratio (MCPR) Safety Limit. The new SR is currently being analyzed for its effect on the MCPR operating limit reported in the Core Operating Limits Report (COLR). A review of the COLR will be performed prior to the approval of this amendment. The requirement in present SR to perform evaluations after completion of control rod drive scram tests is being deleted since the SR as proposed require, through their performance, evaluations of control rod drive scram tests. The requirement to submit the results of the scram time tests in the annual operating report to the NRC is beyond the normal technical specification reporting requirements. Additionally, there are no plant specific design differences that could provide a basis for the additional reporting. Therefore, the requirement is being deleted. Present SR to determine the cycle cumulative mean scram time is being moved to proposed specification 3/4.11, Power Distribution Limits.

<u>Section 3/4.3.E, Control Rod Average Scram Insertion Times</u>

ATTACHMENT 2

This section is a complete adoption of the STS. Proposed Specification 3/4.3.E on Control Rod Average Scram Insertion Times is written from STS guidelines. Proposed LCO 3.3.E requires the average scram insertion time of all operable control rods from the fully withdrawn position based on deenergization of the scram pilot valve solenoids as time zero, to meet the specified limits. The proposed Applicability is operational modes 1 and 2 in order to ensure control rod insertion times are adequate for power operating conditions. The proposed action for 3.3.E requires with the average scram insertion time exceeding any of the limits, the reactor be in at least hot shutdown within 12 hours. The SR for Specification 4.3.E.1 reference the scram time testing requirements of Specification 4.3.D as discussed above.

Section 3/4.3.F, Four Control Rod Group Scram Insertion Times

This section is a complete adoption of the STS with the clarification of slow control rods discussed above. Proposed Specification 3/4.3.F provides requirements for Four Control Rod Group Scram Insertion Times based on the provisions outlined in the STS. The proposed LCO requires that the average of the scram insertion times for the three fastest control rods of all groups of four control rods in a two by two array shall meet the stated limits. The Applicability is operational modes 1 and 2 in order to require compliance in those reactor power operating conditions where control rod scram time assumptions are needed. Proposed action 3.3.F.1 requires the control rods with slower than average scram insertion times to be declared inoperable until an analysis is performed to determine that required scram reactivity remains for the slow four control rod group. Action 3.3.F.2 requires performance of scram time testing in accordance with SR 4.3.D.1.c at least once per 60 days when operation is continued with an average scram insertion time in excess of the specified limit. If the provisions of proposed action 3.3.F.1 are not met, then the reactor is required to be in at least hot shutdown within the next 12 hours. The Surveillance Requirements of Specification 4.3.F reference the provisions of SR 4.3.D for control rod scram time testing.

Section 3/4.3.G, Control Rod Scram Accumulators

This section is a complete adoption of the STS with one additional and one exception to the SRs. This section describes the rewrite of present Specifications for Control Rod Scram Accumulators into proposed Specification 3/4.3.G based on STS provisions. The proposed LCO requires all control rod scram accumulators to be operable with Applicability of operational modes 1, 2, and 5. An exception is provided for operational mode 5 such that the requirements are only for withdrawn control rods and not for control rods that are removed per Specifications 3.10.D or 3.10.E. The present exception, that allows the rod block associated with an inoperable accumulator to be bypassed, is deleted because the proposed specifications require the inoperable accumulator to be made operable within eight hours or the control rod to be fully inserted and declared inoperable. When the control rod is fully inserted and inoperable, the allowance for a control rod block bypass can be removed because the control rod is inoperable and does not need to be included in the specifications. The proposed Applicability covers the operational modes in which control rods can be withdrawn and thus also covers the operating conditions where accumulators may need to perform their intended function. Proposed actions for inoperable control rod scram accumulators do not address the present outdated nine-rod square array requirements, but do implement STS guidelines to help ensure necessary safe plant operation in this condition. Proposed action 3.3.G.1 addresses operational modes 1 or 2 and requires with one control rod scram accumulator inoperable, within 8 hours action be taken to restore the inoperable accumulator to operable status or declare the associated control rod inoperable. If more than one control rod scram accumulator is inoperable in operational modes 1 or 2 control rod drive pump operability must be verified, if the affected control rod(s) is withdrawn, by inserting at least one withdrawn control rod at least one notch or the reactor mode switch must be placed in the Shutdown position. Also, the affected control rods must be inserted and disarmed. If any of these action steps are not met, then the reactor must be in at least hot shutdown within the next 12 hours. A footnote that is used throughout this section is added to this specification and allows for the rearming of the directional control valves to permit testing or returning the control rod to service. In operational mode 5 the action statements require that with one withdrawn control rod with its associated scram accumulator inoperable, the associated control rod be inserted and disarmed. With more than one withdrawn control rod with an associated scram accumulator inoperable in operational mode 5, the reactor mode switch is required to be placed in the Shutdown position. The proposed action statements offer some flexibility over present outdated nine-rod square array provisions but still provide adequate controls to ensure that the control rod scram function is not compromised. The present SR that require checking at least once per shift the status of pressure and level alarms, are replaced with weekly STS testing guidelines. Proposed SR 4.3.G.1 requires at least once per 7 days that the pressure for each control rod scram accumulator be verified to be $\geq$ 800 psig, unless the control rod is inserted and disarmed or scrammed. The one SR that is not adopted from the STS is the channel calibration of the pressure detectors. This surveillance is currently controlled by Station procedures.

Section 3/4.3.H, Control Rod Drive Coupling

This section is a complete adoption of the STS with one clarification. Present specifications imply control rod coupling is required when control rods can be withdrawn. The proposed LCO follows the STS provisions by requiring all control rods to be coupled to their drive mechanisms in Applicable operational modes where control rods can be withdrawn; i.e., operational modes 1, 2 and

5. An exception is provided for operational mode 5 such that only control rods that are withdrawn are required to have coupling integrity and not control rods removed per Specifications 3.10.D or 3.10.E. Proposed action steps for uncoupled control rods follow STS guidelines and provide separate provisions for power operation (operational modes 1 or 2) and for refueling (operational mode 5). Proposed actions with one control rod uncoupled in operational modes 1 or 2 require that within 2 hours, if permitted by the RWM, that the control rod be inserted to attempt and to verify recoupling. If recoupling is not accomplished or, if not permitted by the RWM, then until permitted by the RWM, the control rod is declared inoperable, inserted and disarmed electrically or hydraulically. If neither of the above actions are met, then the reactor must be in at least hot shutdown within 12 hours. A clarification is made to the STS action by deleting the shutdown requirement. This clarification will allow the use of the next action requirement which is enacted when the provisions of the first action requirement is not met. Proposed action 3.3.H.2 is the STS otherwise provision requiring a reactor shutdown within 12 hours if the actions in proposed specification 3.3.H.1 are not met. Proposed action 3.3.H.3 addresses an uncoupled control rod in operational mode 5. The proposed action steps allow within 2 hours either to attempt and to verify recoupling or to insert and disarm the associated control rod. The allowed recoupling is verified by observation of any nuclear instrumentation response and; additionally, by demonstration that the control rod will not go to the overtravel position. Present Surveillance Requirements are changed to match STS guidelines and format. Present coupling checks after each refueling or maintenance that could affect coupling integrity are retained in accordance with STS guidelines. Added to present testing provisions is the requirement to verify each control rod does not go to the overtravel position anytime a control rod is withdrawn to the "Full out" position. The proposed SRs maintain the intent of present requirements by providing demonstrated testing provisions to help ensure control rod drive coupling integrity.

<u>Section 3/4.3.I, Control Rod Position Indication System</u>

This section describes the rewrite of Quad Cities present specifications and is a new specification for Dresden Station. The proposed specifications are a complete adoption of the STS except for 2 differences. Proposed LCO 3.3.I requires all control rod position indicators to be operable in applicable operational modes in accordance with STS guidelines. The proposed Applicability is operational modes 1, 2, and 5 with the provision that applicability for operational mode 5 is only for control rods withdrawn and not for rods removed per Specifications 3.10.D or 3.10.E. The proposed Applicability covers all operational modes in which control rods are withdrawn and in which position indication is needed. Proposed actions for inoperable control rod position indicators are based on STS guidelines and take into consideration the differences between position indication needs at power (operational modes 1 or 2) and during refueling (operational mode 5). Proposed action 3.3.I.1 for operational modes 1 or 2 requires that with one or more position indicators inoperable, the position of the rod be determined by an alternate method, or the rod is moved to a position with an operable indicator, or the rod is declared inoperable, fully inserted and disarmed either electrically or hydraulically. If none of the provisions of proposed action 3.3.I.1 are met, the reactor is required to be in at least hot shutdown within the next 12 hours, proposed specification 3.3.I.2. The proposed action steps implement the intent of the present Quad Cities Specification 3.3.A.3.b by requiring the control rod to be moved to a known position or fully inserted and considered inoperable. A provision from STS is added to allow rearming control rod directional control valves, under administrative control, to allow for testing associated with returning the control rod to operable status. Proposed action 3.3.I.3 addresses operational mode 5 and implements STS guidelines to move the control rod to a known position or insert the control rod. The action steps for operational mode 5 are less stringent than for operational modes 1 or 2 due to rod withdrawal limitations associated with the interlocks with the operational mode. Present Quad Cities Surveillance Requirement 4.3.A.3.a requires that once a shift during power operation and during control rod withdrawal, the control rod display shall be observed for control rod position indication. The intent of the present SR is to ensure that position indication is determined on a frequency that will help to ensure operability of the system. The intent of this present SR is implemented by proposed SRs 4.4.I.1, 2, and 3 which are based on the STS. The proposed SRs require checking that the position of each control rod is indicated at least once per 24 hours, during movement of control rods for the notch movement tests of SR 4.3.C.1 and the "Full out" position test of SR 4.3.H.2. The 2 differences between the proposed specification and the STS specifications are as follow:

1) STS provides separate actions for inoperable "Full-in" or "Full-out" position indicators that are redundant with action requirements for normal position indicators. These are treated the same at Dresden and Quad Cities and therefore a unique action was not identified in the proposed actions.

2) STS provides separate actions when thermal power is above or below the low power setpoint of the Rod Sequence Control System. Dresden and Quad Cities do not have a Rod Sequence Control System and therefore, the required actions are not based upon thermal power.

<u>Section 3/4.3.J, Control Rod Drive Housing Support</u>

This section is a complete adoption of the STS. This section details the rewrite of the present specification on Control Rod Drive Housing Supports to proposed Specification 3/4.3.J. The proposed LCO requires the control rod drive housing supports to be in place with an Applicability of operational modes 1, 2, and 3. The proposed LCO and Applicability follow STS guidelines and implement STS provisions. Present provisions require the control rod drive housing support to be in place during reactor power operation and when the reactor coolant system is pressurized above atmospheric pressure with fuel in the vessel, unless all control rods are fully inserted and the shutdown margin provisions are met. The proposed applicability of operational modes 1, 2, and 3 cover the reactor conditions of power operation and when the reactor coolant system can be pressurized. Present Specifications contains no action steps when the control rod drive housing supports are not in place in the applicable operational modes. The proposed action follows STS guidelines by requiring the reactor to be in at least hot shutdown within 12 hours and in cold shutdown within the following 24 hours. The present Surveillance Requirement for the control rod drive housing supports is rewritten using STS wording as SR 4.3.J. The rewrite is more explicit by requiring a visual inspection prior to startup any time the control rod drive housing support has been disassembled or maintenance has been performed in the control rod drive housing support area.

Section 3/4.3.K, Scram Discharge Volume Vent and Drain Valves

This section is a spinoff of STS section 3.1.3.1. The LCO and applicability are based on STS section 3.1.3.1. The actions and surveillance requirements are adopted from STS with two exceptions in the SR. The section is proposed to be contained within its own section which is different than STS. The present specifications for the scram discharge volume vent and drain valves address only the Surveillance Requirements and do not contain LCO, Applicability or action provisions. The proposed rewrite uses the provisions from the STS to address the missing requirements in the Dresden and Quad Cities Technical Specifications. Proposed Specifications 3.3.K/4.3.K address the operability of the scram discharge volume vent and drain valves. The proposed LCO requires all scram discharge volume vent and drain valves to be operable with an Applicability of operational modes 1 and 2. Operational modes 1 and 2 were chosen since these are the only conditions where multiple rod scrams are needed and thus the operability of the affected vent and drain valves are required to ensure system integrity. Proposed action 3.3.K.1 addresses the condition where one scram discharge volume vent valve or one scram discharge volume drain valve is inoperable and open; or, a combination of any one scram discharge volume vent valve and any one drain valve are inoperable and open. The valve(s) are required to be restored to operable status within 24 hours or the reactor shall be in at least hot shutdown within the next 12 hours. Proposed action 3.3.K.2 addresses all other possible combinations of vent and drain valve inoperability other than those addressed in Proposed action 3.3.K.1 and requires the valve(s) to be returned to operable status within 8 hours or the reactor shall be in at least hot shutdown within the next 12 hours. The proposed action steps address necessary conditions of inoperability for the scram discharge volume vent and drain valves and provide reasonable out of service times for repairs. Surveillance Requirements for the scram discharge volume vent and drain valves are adopted from the STS except for two. Proposed SR 4.3.K.3 is not explicitly tied to control rod testing and control rod density. The SR for the functional test of the scram discharge volume level sensors are not adopted. These surveillances already reside in the TS in sections 3/4.1 and 3/4.2. Therefore, the SR will not be added to section 3/4.3.

Section 3/4.3.L. Rod Worth Minimizer.

This section is a complete adoption of the STS. Present Specifications describe how control rod sequences shall be established to limit maximum reactivity addition due to control rod dropout so that the rod drop accident design limit of 280 cal/gm is not exceeded. This information should not be contained in the technical specifications but is more appropriately located in the Bases. Present Specifications are rewritten using STS guidelines as Specifications 3/4.3.L. The proposed LCO requires the RWM to be operable with Applicability of operational modes 1 and 2, when thermal power is less than 20% (10% for Quad Cities) of rated thermal power. Proposed action steps for an inoperable RWM are taken from STS guidelines and allow a second licensed operator or technically qualified individual who is present at the reactor control console to verify rod movement and compliance with the prescribed control rod pattern. Otherwise, control rod movement is allowed only by actuating a reactor scram or placing the reactor mode switch in the Shutdown position. The proposed action steps are different from present provisions since presently, at least 12 control rods must be fully withdrawn before a second operator or qualified technical person can be used as a substitute for the RWM. The proposed action has been generically determined to provide adequate assurance that control rods will be withdrawn in accordance with prescribed patterns (without the necessity of requiring 12 control rods to be fully withdrawn before a substitute can be used for the RWM). Proposed Surveillance Requirements 4.3.L for the Rod Worth Minimizer are taken from STS guidelines. Verification of the selection error of at least one out-of-sequence control rod is performed in operational mode 1 prior to reducing thermal power below 20% (10% for Quad Cities) and in operational mode 2 within 8 hours prior to withdrawal of control rods to make the reactor critical. Testing to demonstrate the inability to withdraw an out of sequence control rod is performed in operational mode 1 prior to reducing thermal power below 20% (10% for Quad Cities) rated thermal power and in operational mode 2 within 8 hours prior to withdrawal of control rods to make the reactor critical. The RWM at Dresden and Quad Cities is normally active at all reactor power levels and thus testing can be performed at > 20% (10% for Quad Cities) rated thermal power. Proposed SR 4.3.L.1 provides demonstration that the control rod patterns and sequence input to the RWM computer are correctly loaded following any loading of the program into the computer. The proposed low power setpoint of 10% for Quad Cities is based on "NRC Safety Evaluation Report Approving Amendment 17 to NEDE-24011-P Dated December 27, 1987." The safety evaluation was issued in response to a topical report submitted by the BWR Owners Group. The topical report proposed changes to the Rod Sequence Control System (Dresden nor Quad Cities have this system) and the lowering of the low power setpoint from 20% to 10%. The NRC found the report acceptable for referencing in license applications. Quad Cities neutronic analyses are currently performed in accordance with NEDE-24011-P and therefore the topical report is applicable. Siemens Nuclear currently performs the neutronic analysis for Dresden Station and thus the topical report is currently not applicable and therefore, the lowering of the low power setpoint is not requested for Dresden.

Section 3/4.3.M, Rod Block Monitor

This section is a complete adoption of the STS. This section describes the rewrite of present specifications for the Rod Block Monitor to proposed Specification 3/4.3.M, titled Rod Block Monitor. The proposed LCO implements present requirements by requiring both RBM channels to be operable. However, the present provision of requiring the RBM operable only during limiting control rod patterns is replaced with the STS Applicability provision of operational mode 1, when thermal power is greater than or equal to 30% of rated thermal power. Proposed action steps for the RBM are based on STS guidelines. Action 3.3.M.1 requires that with one RBM channel inoperable, verification be made that the reactor is not operating in a limiting control rod pattern and that the inoperable RBM channel be restored to operable status within 24 hours. If these action provisions are not met, then the inoperable RBM is required to be in the tripped condition within one hour. Proposed action 3.3.M.2 addresses the condition where both RBM channels are inoperable and requires that at least one be in the tripped condition within one hour. Present requirements do not contain time frames for accomplishing action steps and; thus, STS time frames are utilized. The Surveillance Requirements for the RBM channel are taken from STS requirements. SR 4.3.M.1 references Table 4.2.E-1 in order to avoid duplicating the SR listing. Proposed SR 4.3.M.2 requires a channel functional test prior to control rod withdrawal when the reactor is operating in a limiting control rod pattern. A clarification was added to the SR for the channel functional test to be performed when the reactor is operating in a limiting control rod pattern but no more than daily. The STS requires the functional test to be performed when the reactor is operating in a limiting control rod pattern and daily thereafter. The STS intent was to perform the channel functional test prior to withdrawing control rods when the reactor is operating in a limiting control rod pattern but not more than daily.

Section 3/4.3.N, Economic Generation Control (EGC) System

ATTACHMENT 2

Proposed Specification 3/4.3.N on the Economic Generation Control System retains the present provisions for EGC operation. STS format and wording style are used for consistency with other proposed changes.

The proposed changes to the Limiting Conditions for Operation Bases 3.3 are made to support the changes proposed to the individual specifications. The proposed Bases are a combination of the STS Bases, the current Dresden Bases, the current Quad Cities Bases and include addtional information as determined necessary.

ATTACHMENT 3

PROPOSED TECHNICAL SPECIFICATIONS

Technical Specification 3/4.3

"REACTIVITY CONTROL"

3.3 - LIMITING CONDITIONS FOR OPERATION

A. SHUTDOWN MARGIN (SDM)

The SHUTDOWN MARGIN (SDM) shall be equal to or greater than:

1. 0.35% $\Delta k/k$ with the highest worth control rod analytically determined, or

2. 0.25% $\Delta k/k$ with the highest worth control rod determined by test.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, 3, 4, and 5.

ACTION:

With the SHUTDOWN MARGIN less than specified:

1. In OPERATIONAL MODE 1 or 2, restore the required SHUTDOWN MARGIN within 6 hours or be in at least HOT SHUTDOWN within the next 12 hours.

2. In OPERATIONAL MODE 3 or 4, immediately verify all insertable control rods to be fully inserted and suspend all activities that could reduce the SHUTDOWN MARGIN. In OPERATIONAL MODE 4, establish SECONDARY CONTAINMENT INTEGRITY within 8 hours.

3. In OPERATIONAL MODE 5, suspend CORE ALTERATION(s) and other activities that could reduce the SHUTDOWN MARGIN and fully insert all insertable control rods within 1 hour. Establish SECONDARY CONTAINMENT INTEGRITY within 8 hours.

4.3 - SURVEILLANCE REQUIREMENTS

A. SHUTDOWN MARGIN

The SHUTDOWN MARGIN shall be determined to be equal to or greater than that specified at any time during the operating cycle:

1. By demonstration, prior to or during the first startup after each refueling outage.

2. Within 24 hours after detection of a withdrawn control rod that is immovable, as a result of excessive friction or mechanical interference, or known to be unscrammable. The required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or unscrammable control rod.

3. By calculation, prior to each fuel movement during the fuel loading sequence.

3.3 - LIMITING CONDITIONS FOR OPERATION

B. Reactivity Anomalies

The reactivity equivalence of the difference between the actual critical control rod configuration and the predicted control rod configuration shall not exceed 1% Δk/k.

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2.

ACTION:

With the reactivity equivalence difference exceeding 1% Δk/k, within 12 hours perform an analysis to determine and explain the cause of the reactivity difference; operation may continue if the difference is explained and corrected.

With the provisions of the ACTION above not met, be in at least HOT SHUTDOWN within the next 12 hours.

4.3 - SURVEILLANCE REQUIREMENTS

B. Reactivity Anomalies

The reactivity equivalence of the difference between the actual critical control rod configuration and the predicted control rod configuration shall be verified to be less than or equal to 1% Δk/k:

1. During the first startup following CORE ALTERATION(s), and

2. At least once per 31 effective full power days.

3.3 - LIMITING CONDITIONS FOR OPERATION

C. Control Rod OPERABILITY

All control rods shall be OPERABLE.

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2.

ACTION:

1. With one control rod inoperable due to being immovable as a result of excessive friction or mechanical interference, or known to be unscrammable:
    a. Within one hour:
        1) Verify that the inoperable control rod, if withdrawn, is separated from all other inoperable withdrawn control rods by at least two control cells in all directions.
        2) Disarm the associated directional control valves[a] either:
            a) Electrically, or
            b) Hydraulically by closing the drive water and exhaust water isolation valves.
    b. With the provisions of ACTION 1.a above not met, be in at least HOT SHUTDOWN within the next 12 hours.

4.3 - SURVEILLANCE REQUIREMENTS

C. Control Rod OPERABILITY

1. When above the low power setpoint of the RWM, all withdrawn control rods not required to have their directional control valves disarmed electrically or hydraulically shall be demonstrated OPERABLE by moving each control rod at least one notch:
    a. At least once per 7 days, and
    b. At least once per 24 hours when any control rod is immovable as a result of excessive friction or mechanical interference, or known to be unscrammable.
2. All control rods shall be demonstrated OPERABLE by performance of Surveillance Requirements 4.3.D, 4.3.G, 4.3.H and 4.3.I.

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a May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.

3.3 - LIMITING CONDITIONS FOR OPERATION

4.3 - SURVEILLANCE REQUIREMENTS

c. Comply with Surveillance Requirement 4.3.A.2 within 24 hours or be in HOT SHUTDOWN within the next 12 hours.

2. With one or more control rods scrammable but inoperable for causes other than addressed in ACTION 3.3.C.1 above:

a. If the inoperable control rod(s) is withdrawn, within one hour:

1) Verify that the inoperable withdrawn control rod(s) is separated from all other inoperable withdrawn control rods by at least two control cells in all directions, and

2) Demonstrate the insertion capability of the inoperable withdrawn control rod(s) by inserting the inoperable withdrawn control rod(s) at least one notch by drive water pressure within the normal operating range.[b]

b. With the provisions of ACTION 2.a above not met, fully insert the inoperable withdrawn control rod(s) and disarm the associated directional control valves[a] either:

1) Electrically, or

2) Hydraulically by closing the drive water and exhaust water isolation valves.

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b The inoperable control rod may then be withdrawn to a position no further withdrawn than its position when found to be inoperable.

a May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.

3.3 - LIMITING CONDITIONS FOR OPERATION

4.3 - SURVEILLANCE REQUIREMENTS

c. If the inoperable control rod(s) is fully inserted, within one hour disarm the associated directional control valves[a] either:

1) Electrically, or

2) Hydraulically by closing the drive water and exhaust water isolation valves.

3. With the provisions of ACTION 2 above not met, be in at least HOT SHUTDOWN within the next 12 hours.

4. With more than 8 control rods inoperable, be in at least HOT SHUTDOWN within 12 hours.

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a May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.

3.3 - LIMITING CONDITIONS FOR OPERATION

D. Maximum Scram Insertion Times

The maximum scram insertion time of each control rod from the fully withdrawn position to 90% insertion, based on de-energization of the scram pilot valve solenoids as time zero, shall not exceed 7 seconds.

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2.

ACTION:

With the maximum scram insertion time of one or more control rods exceeding 7 seconds:

1. Declare the control rod(s) exceeding the above maximum scram insertion time inoperable, and
2. When operation is continued with three or more control rods with maximum scram insertion times in excess of 7 seconds, perform Surveillance Requirement 4.3.D.3 at least once per 60 days of POWER OPERATION.

With the provisions of the ACTION(s) above not met, be in at least HOT SHUTDOWN within 12 hours.

4.3 - SURVEILLANCE REQUIREMENTS

D. Maximum Scram Insertion Times

The maximum scram insertion time of the control rods shall be demonstrated through measurement with reactor coolant pressure greater than 800 psig and, during single control rod scram time tests, with the control rod drive pumps isolated from the accumulators:

1. For all control rods prior to THERMAL POWER exceeding 40% of RATED THERMAL POWER:
   a. following CORE ALTERATION(s), or
   b. after a reactor shutdown that is greater than 120 days,
2. For specifically affected individual control rods[a] following maintenance on or modification to the control rod or control rod drive system which could affect the scram insertion time of those specific control rods, and
3. For at least 10% of the control rods, on a rotating basis, at least once per 120 days of POWER OPERATION.

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a The provisions of Specification 4.0.D are not applicable provided this surveillance is conducted prior to exceeding 40% of RATED THERMAL POWER.

3.3 - LIMITING CONDITIONS FOR OPERATION

E. Average Scram Insertion Times

The average scram insertion time of all OPERABLE control rods from the fully withdrawn position, based on de-energization of the scram pilot valve solenoids as time zero, shall not exceed any of the following:

% Inserted From Fully Withdrawn	Avg. Scram Insertion Times (sec)
5	0.375
20	0.900
50	2.00
90	3.50

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2.

ACTION:

With the average scram insertion time exceeding any of the above limits, be in at least HOT SHUTDOWN within 12 hours.

4.3 - SURVEILLANCE REQUIREMENTS

E. Average Scram Insertion Times

The control rod average scram times shall be demonstrated by scram time testing from the fully withdrawn position as required by Surveillance Requirement 4.3.D.

3.3 - LIMITING CONDITIONS FOR OPERATION

F. Group Scram Insertion Times

The average of the scram insertion times, from the fully withdrawn position, for the three fastest control rods of all groups of four control rods in a two-by-two array, based on de-energization of the scram pilot valve solenoids as time zero, shall not exceed any of the following:

% Inserted From Fully Withdrawn	Avg. Scram Insertion Times (sec)
5	0.398
20	0.954
50	2.120
90	3.800

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2.

ACTION:

With the average scram insertion times of control rods exceeding the above limits:

1. Declare the control rods exceeding the above average scram insertion times inoperable until an analysis is performed to determine that required scram reactivity remains for the slow four control rod group, and
2. When operation is continued with an average scram insertion time(s) in excess of the average scram insertion time limit, perform Surveillance Requirement 4.3.D.3 at least once per 60 days of POWER OPERATION.

With the provisions of the ACTION(s) above not met, be in at least HOT SHUTDOWN within 12 hours.

4.3 - SURVEILLANCE REQUIREMENTS

F. Group Scram Insertion Times

All control rods shall be demonstrated OPERABLE by scram time testing from the fully withdrawn position as required by Surveillance Requirement 4.3.D.

3.3 - LIMITING CONDITIONS FOR OPERATION

G. Control Rod Scram Accumulators

All control rod scram accumulators shall be OPERABLE.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 5[a].

ACTION:

1. In OPERATIONAL MODE 1 or 2:

   a. With one control rod scram accumulator inoperable, within 8 hours:

      1) Restore the inoperable accumulator to OPERABLE status, or

      2) Declare the control rod associated with the inoperable accumulator inoperable.

   b. With the provisions of ACTION 1.a above not met, be in at least HOT SHUTDOWN within the next 12 hours.

   c. With more than one control rod scram accumulator inoperable, declare the associated control rods inoperable and:

4.3 - SURVEILLANCE REQUIREMENTS

G. Control Rod Scram Accumulators

Each control rod scram accumulator shall be determined OPERABLE at least once per 7 days by verifying that the indicated pressure is $\geq$800 psig unless the control rod is fully inserted and disarmed, or scrammed.

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a In OPERATIONAL MODE 5, this Specification is applicable for the accumulators associated with each withdrawn control rod and is not applicable to control rods removed per Specification 3.10.D or 3.10.E.

3.3 - LIMITING CONDITIONS FOR OPERATION

1) If the control rod associated with any inoperable scram accumulator is withdrawn, immediately verify that at least one control rod drive pump is operating by inserting at least one withdrawn control rod at least one notch. With no control rod drive pump operating, immediately place the reactor mode switch in the Shutdown position.

2) Fully insert the inoperable control rods and disarm the associated directional control valves[b] either:

a) Electrically, or

b) Hydraulically by closing the drive water and exhaust water isolation valves.

d. With the provisions of ACTION 1.c.2 above not met, be in at least HOT SHUTDOWN within 12 hours.

2. In OPERATIONAL MODE 5[a]:

a. With one withdrawn control rod with its associated scram accumulator inoperable, fully insert the affected control rod and disarm the associated directional control valves[b] within one hour, either:

4.3 - SURVEILLANCE REQUIREMENTS

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a In OPERATIONAL MODE 5, this Specification is applicable for the accumulators associated with each withdrawn control rod and is not applicable to control rods removed per Specification 3.10.D or 3.10.E.

b May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.

3.3 - LIMITING CONDITIONS FOR OPERATION

1) Electrically, or

2) Hydraulically by closing the drive water and exhaust water isolation valves.

b. With more than one withdrawn control rod with the associated scram accumulator inoperable or no control rod drive pump operating, immediately place the reactor mode switch in the Shutdown position.

4.3 - SURVEILLANCE REQUIREMENTS

3.3 - LIMITING CONDITIONS FOR OPERATION

H. Control Rod Drive Coupling

All control rods shall be coupled to their drive mechanisms.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, and 5[a].

ACTION:

1. In OPERATIONAL MODE 1 or 2 with one control rod not coupled to its associated drive mechanism, within 2 hours:
   a. If permitted by the RWM, insert the control rod drive mechanism to accomplish recoupling and verify recoupling by withdrawing the control rod, and:
      1) Observing any indicated response of the nuclear instrumentation, and
      2) Demonstrating that the control rod will not go to the overtravel position.
   b. If not permitted by the RWM or, if recoupling is not accomplished in accordance with ACTION 1.a above, then declare the control rod inoperable, fully insert the control rod and disarm the associated directional control valves[b] either:
      1) Electrically, or

4.3 - SURVEILLANCE REQUIREMENTS

H. Control Rod Drive Coupling

Each affected control rod shall be demonstrated to be coupled to its drive mechanism by observing any indicated response of the nuclear instrumentation while withdrawing the control rod to the fully withdrawn position and then verifying that the control rod drive does not go to the overtravel position:

1. Prior to reactor criticality after completing CORE ALTERATION(s) that could have affected the control rod drive coupling integrity,
2. Anytime the control rod is withdrawn to the "Full out" position in subsequent operation, and
3. Following maintenance on or modification to the control rod or control rod drive system which could have affected the control rod drive coupling integrity.

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a In OPERATIONAL MODE 5, this Specification is applicable for withdrawn control rods and is not applicable to control rods removed per Specification 3.10.D or 3.10.E.

b May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.

3.3 - LIMITING CONDITIONS FOR OPERATION

2) Hydraulically by closing the drive water and exhaust water isolation valves.

2. With the provisions of ACTION 1 above not met, be in at least HOT SHUTDOWN within 12 hours.

3. In OPERATIONAL MODE 5[a] with a withdrawn control rod not coupled to its associated drive mechanism, within 2 hours:

   a. Insert the control rod to accomplish recoupling and verify recoupling by withdrawing control rod and demonstrating that the control rod will not go to the overtravel position, or

   b. If recoupling is not accomplished, declare the control rod inoperable, fully insert the control rod and disarm the associated directional control valves[b] within one hour, either:

      1) Electrically, or

      2) Hydraulically by closing the drive water and exhaust water isolation valves.

4.3 - SURVEILLANCE REQUIREMENTS

a In OPERATIONAL MODE 5, this Specification is applicable for withdrawn control rod and is not applicable to control rods removed per Specification 3.10.D or 3.10.E.

b May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.

3.3 - LIMITING CONDITIONS FOR OPERATION

I. Control Rod Position Indication System

All control rod position indicators shall be OPERABLE.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, and 5[a].

ACTION:

1. In OPERATIONAL MODE 1 or 2 with one or more control rod position indicators inoperable, within one hour either:
   a. Determine the position of the control rod by an alternate method, or
   b. Move the control rod to a position with an OPERABLE position indicator, or
   c. Declare the control rod inoperable, fully insert the inoperable withdrawn control rod(s), and disarm the associated directional control valves[b] either:
      1) Electrically, or
      2) Hydraulically by closing the drive water and exhaust water isolation valves.

4.3 - SURVEILLANCE REQUIREMENTS

I. Control Rod Position Indication System

The control rod position indication system shall be determined OPERABLE by verifying:

1. At least once per 24 hours that the position of each control rod is indicated.
2. That the indicated control rod position changes during the movement of the control rod drive when performing Surveillance Requirement 4.3.C.1.
3. That the control rod position indicator corresponds to the control rod position indicated by the "Full out" position indicator when performing Surveillance Requirement 4.3.H.2.

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a In OPERATIONAL MODE 5, this Specification is applicable for withdrawn control rods and is not applicable to control rods removed per Specification 3.10.D or 3.10.E.

b May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod(s) to OPERABLE status.

3.3 - LIMITING CONDITIONS FOR OPERATION

2. With the provisions of ACTION 1 above not met, be in at least HOT SHUTDOWN within the next 12 hours.

3. In OPERATIONAL MODE 5[a] with a withdrawn control rod position indicator inoperable:

   a. Move the control rod to a position with an OPERABLE position indicator, or

   b. Fully insert the control rod.

4.3 - SURVEILLANCE REQUIREMENTS

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a In OPERATIONAL MODE 5, this Specification is applicable for withdrawn control rods and is not applicable to control rods removed per Specification 3.10.D or 3.10.E.

3.3 - LIMITING CONDITIONS FOR OPERATION

J. Control Rod Drive Housing Support

The control rod drive housing support shall be in place.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, and 3.

ACTION:

With the control rod drive housing support not in place, be in at least HOT SHUTDOWN within 12 hours and in at least COLD SHUTDOWN within the following 24 hours.

4.3 - SURVEILLANCE REQUIREMENTS

J. Control Rod Drive Housing Support

The control rod drive housing support shall be verified to be in place by a visual inspection prior to startup any time it has been disassembled or when maintenance has been performed in the control rod drive housing support area.

3.3 - LIMITING CONDITIONS FOR OPERATION

K. SDV Vent and Drain Valves

All scram discharge volume (SDV) vent and drain valves shall be OPERABLE.

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2.

ACTION:

1. With one scram discharge volume vent valve and/or one scram discharge volume drain valve inoperable and open, restore the inoperable valve(s) to OPERABLE status within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours.

2. With any scram discharge volume vent valve(s) and/or any scram discharge volume drain valve(s) otherwise inoperable, restore at least one vent valve and one drain valve to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours.

4.3 - SURVEILLANCE REQUIREMENTS

K. SDV Vent and Drain Valves

The scram discharge volume vent and drain valves shall be demonstrated OPERABLE:

1. At least once per 31 days by verifying each valve to be open[a], and

2. At least once per 92 days by cycling each valve through at least one complete cycle of travel.

3. At least once per 18 months, the scram discharge volume vent and drain valves shall be demonstrated to:

   a. Close within 30 seconds after receipt of a signal for control rods to scram, and

   b. Open after the scram signal is reset.

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a These valves may be closed intermittently for testing under administrative controls.

3.3 - LIMITING CONDITIONS FOR OPERATION

L. Rod Worth Minimizer (RWM)

The rod worth minimizer (RWM) shall be OPERABLE.

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2[a], when THERMAL POWER is less than or equal to 20% of RATED THERMAL POWER.

ACTION:

With the RWM inoperable, verify control rod movement and compliance with the prescribed control rod pattern by a second licensed operator or technically qualified individual who is present at the reactor control console. Otherwise, control rod movement may be made only by actuating the manual scram or placing the reactor mode switch in the Shutdown position.

4.3 - SURVEILLANCE REQUIREMENTS

L. Rod Worth Minimizer (RWM)

The RWM shall be demonstrated OPERABLE:

1. By verifying that the control rod patterns and sequence input to the RWM computer are correctly loaded following any loading of the program into the computer.

2. In OPERATIONAL MODE 2 within 8 hours prior to withdrawal of control rods for the purpose of making the reactor critical:

   a. by verifying proper indication of the selection error of at least one out-of-sequence control rod.

   b. by verifying the rod block function by demonstrating inability to withdraw an out-of-sequence control rod.

3. In OPERATIONAL MODE 1 prior to reducing THERMAL POWER below 20% of RATED THERMAL POWER:

   a. by verifying proper indication of the selection error of at least one out-of-sequence control rod.

   b. by verifying the rod block function by demonstrating inability to withdraw an out-of-sequence control rod.

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a Entry into OPERATIONAL MODE 2 and withdrawal of selected control rods is permitted for the purpose of determining the OPERABILITY of the RWM prior to withdrawal of control rods for the purpose of bringing the reactor to criticality.

3.3 - LIMITING CONDITIONS FOR OPERATION

M. Rod Block Monitor (RBM)

Both rod block monitor (RBM) CHANNEL(s) shall be OPERABLE.

APPLICABILITY:

OPERATIONAL MODE 1, when thermal power is greater than or equal to 30% of RATED THERMAL POWER.

ACTION:

1. With one RBM CHANNEL inoperable:
   a. Verify that the reactor is not operating in a LIMITING CONTROL ROD PATTERN, and
   b. Restore the inoperable RBM CHANNEL to OPERABLE status within 24 hours.
2. With the provisions of ACTION 1 above not met, place the inoperable rod block monitor CHANNEL in the tripped condition within the next one hour.
3. With both RBM CHANNEL(s) inoperable, place at least one inoperable rod block monitor CHANNEL in the tripped condition within one hour.

4.3 - SURVEILLANCE REQUIREMENTS

M. Rod Block Monitor (RBM)

Each of the required RBM CHANNEL(s) shall be demonstrated OPERABLE by performance of a:

1. CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION at the frequencies and for the OPERATIONAL MODE(s) specified in Table 4.2.E-1.
2. CHANNEL FUNCTIONAL TEST prior to control rod withdrawal when the reactor is operating in a LIMITING CONTROL ROD PATTERN, but no more often than daily.

3.3 - LIMITING CONDITIONS FOR OPERATION

N. Economic Generation Control (EGC) System

The economic generation control (EGC) system may be in operation with automatic flow control provided:

a. Core flow is within 65% to 100% of rated core flow, and

b. THERMAL POWER is ≥ 20% of RATED THERMAL POWER.

APPLICABILITY

OPERATIONAL MODE 1.

ACTION:

With core flow less than 65% or greater than 100% of rated core flow, or THERMAL POWER less than 20% of RATED THERMAL POWER, restore operation to within the limits within one hour. Otherwise, immediately remove the plant from EGC operation.

4.3 - SURVEILLANCE REQUIREMENTS

N. Economic Generation Control (EGC) System

The economic generation control system shall be demonstrated OPERABLE by verifying that core flow is within 65% to 100% of rated core flow and THERMAL POWER is ≥ 20% of RATED THERMAL POWER:

a. Prior to entry into EGC operation, and

b. At least once per 12 hours while operating in EGC.

BASES

3/4.3.A SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

The SHUTDOWN MARGIN limitation is a restriction to be applied principally to a new refueling pattern. Satisfaction of the limitation must be determined at the time of loading and must be such that it will apply to the entire subsequent fuel cycle. This determination is provided by core design calculations and administrative control of fuel loading patterns. These procedures include restrictions to allow only those intermediate fuel assembly configurations that have been shown to provide the required SHUTDOWN MARGIN.

Since core reactivity values will vary through core life as a function of fuel depletion and poison burnup, the demonstration of SHUTDOWN MARGIN will be performed during xenon free conditions and adjusted to 68°F to accommodate the current moderator temperature. The generalized form is that the reactivity of the core loading will be limited so the core can be made subcritical by at least R + 0.35% Δk/k or R + 0.25% Δk/k, as appropriate, with the strongest control rod fully withdrawn and all others fully inserted. Two different values are supplied in the Limiting Condition for Operation to provide for the different methods of determination of the highest control rod worth, either analytically or by test. This is due to the reduced uncertainty in the SHUTDOWN MARGIN test when the highest worth control rod is determined by demonstration. When SHUTDOWN MARGIN is determined by calculations not associated with a test, additional margin must be added to the specified SHUTDOWN MARGIN limit to account for uncertainties in the calculation.

The value of R in units of % Δk/k is the difference between the calculated beginning-of-life core reactivity, at the beginning of the operating cycle, and the calculated value of maximum core reactivity at any time later in the operating cycle, where it would be greater than at the beginning. The value of R shall include the potential SHUTDOWN MARGIN loss assuming full B_4C settling in all inverted poison tubes present in the core. R must be a positive quantity or zero and a new value of R must be determined for each new fuel cycle.

The value of % Δk/k in the above expression is provided as a finite, demonstrable, subcriticality margin. This margin may be demonstrated by full withdrawal of the strongest rod and partial withdrawal of an adjacent rod to a position calculated to add at least R + 0.25% Δk/k (or 0.35% Δk/k) in reactivity. Observation of subcriticality in this condition assures subcriticality with not only the strongest rod fully withdrawn but with a substantial margin beyond this condition. This reactivity characteristic has been a basic assumption in the analysis of plant performance and can be best demonstrated at the time of fuel loading, but the margin must also be determined anytime a control rod is incapable of insertion following a scram signal. Any control rod that is immovable as a result of excessive friction or mechanical interference, or is known to be unscrammable, per Specification 3.3.C, is considered to be incapable of insertion following a scram signal. It is important to note that a control rod can be electrically immovable, but scrammable, and no increase in SHUTDOWN MARGIN is required for these control rods.

3/4.3.B Reactivity Anomalies

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

3/4.3.C Control Rod OPERABILITY

Control rods are the primary reactivity control system for the reactor. In conjunction with the Reactor Protection System, the control rods provide the means for reliable control of reactivity changes to ensure the specified acceptable fuel design limits are not exceeded. This specification, along with others, assures that the performance of the control rods in the event of an accident or transient, meets the assumptions used in the safety analysis. Of primary concern is the trippability of the control rods. Other causes for inoperability are addressed in other Specifications following this one. However, the inability to move a control rod which remains trippable does not prevent the performance of the control rod's safety function.

The specification requires that a rod be taken out-of-service if it cannot be moved with drive pressure. Damage within the control rod drive mechanism could be a generic problem, therefore with a control rod immovable because of excessive friction or mechanical interference, operation of the reactor is limited to a time period which is reasonable to determine the cause of the inoperability and at the same time prevent operation with a large number of inoperable control rods. Control rods that are inoperable due to exceeding allowed scram times, but are movable by control rod drive pressure, need not be disarmed electrically if the shutdown margin provisions are met for each position of the affected rod(s).

If the rod is fully inserted and then disarmed electrically or hydraulically, it is in a safe position of maximum contribution to shutdown reactivity. (Note: To disarm the drive electrically, four amphenol-type plug connectors are removed from the drive insert and withdrawal solenoids, rendering the drive immovable. This procedure is equivalent to valving out the drive and is preferred, as drive water cools and minimizes crud accumulation in the drive.). If it is disarmed electrically in a non-fully inserted position, that position shall be consistent with the SHUTDOWN MARGIN limitation stated in Specification 3.3.A. This assures that the core can be shut down at all times with the remaining control rods, assuming the strongest OPERABLE control rod does not insert. The occurrence of more than eight inoperable control rods could be indicative of a generic control rod drive problem which requires prompt investigation and resolution.

In order to reduce the potential for Control Rod Drive (CRD) damage and more specifically, collet housing failure, a program of disassembly and inspection of CRDs is conducted during or after each refueling outage. This program follows the recommendations of General Electric SIL-139 with nondestructive examination results compiled and reported to General Electric on collet housing cracking problems.

The required surveillance intervals are adequate to determine that the rods are OPERABLE and not so frequent as to cause excessive wear on the system components.

<u>3/4.3.D</u> <u>Control Rod Maximum Scram Insertion Times;</u>

<u>3/4.3.E</u> <u>Control Rod Average Scram Insertion Times; and</u>

<u>3/4.3.F</u> <u>Four Control Rod Group Scram Insertion Times</u>

These specifications ensure that the control rod insertion times are consistent with those used in the safety analyses. The control rod system is analyzed to bring the reactor subcritical at a rate fast enough to prevent fuel damage, i.e., to prevent the MCPR from becoming less than the fuel cladding integrity Safety Limit. The analyses demonstrate that if the reactor is operated within the limitation set in Specification 3.11.C, the negative reactivity insertion rates associated with the scram performance (as adjusted for statistical variation in the observed data) result in protection of the MCPR Safety Limit.

Analysis of the limiting power transient shows that the negative reactivity rates, resulting from the scram with the average response of all the drives, as given in the above specification, provide the required protection, and MCPR remains greater than the fuel cladding integrity SAFETY LIMIT. In the analytical treatment of most transients, 290 milliseconds are allowed between a neutron sensor reaching the scram point and the start of motion of the control rods. This is adequate and conservative when compared to the typically observed time delay of about 210 milliseconds. Approximately 90 milliseconds after neutron flux reaches the trip point, the pilot scram valve solenoid de-energizes and 120 milliseconds later the control rod motion is estimated to actually begin. However, 200 milliseconds rather than 120 milliseconds is conservatively assumed for this time interval in the transient analyses and is also included in the allowable scram insertion times specified in Specifications 3.3.D, 3.3.E, and 3.3.F. In the statistical treatment of the limiting transients, a statistical distribution of total scram delay is used rather than the bounding value described above.

BASES

The performance of the individual control rod drives is monitored to assure that scram performance is not degraded. Observed plant data or Technical Specification limits were used to determine the average scram performance used in the transient analyses, and the results of each set of control rod scram tests performed during the current cycle are compared against earlier results to verify that the performance of the control rod insertion system has not changed significantly.

If test results should be determined to fall outside of the statistical population defining the scram performance characteristics used in the transient analyses, a re-determination of thermal margin requirements is undertaken as required by Specification 3.11.C. A smaller test sample than that required by these specifications is not statistically significant and should not be used in the re-determination of thermal margins. Individual control rod drives with excessive scram times can be fully inserted into the core and de-energized in the manner of an inoperable rod drive provided the allowable number of inoperable control rod drives is not exceeded. In this case, the scram speed of the drive shall not be used as a basis in the re-determination of thermal margin requirements. For excessive average scram insertion times, only the individual control rods in the two-by-two array which exceed the allowed average scram insertion time are considered inoperable.

The scram times for all control rods are measured at the time of each refueling outage. Experience with the plant has shown that control drive insertion times vary little through the operating cycle; hence no re-assessment of thermal margin requirements is expected under normal conditions. The history of drive performance accumulated to date indicates that the 90% insertion times of new and overhauled drives approximate a normal distribution about the mean which tends to become skewed toward longer scram times as operating time is accumulated. The probability of a drive not exceeding the mean 90% insertion time by 0.75 seconds is greater than 0.999 for a normal distribution. The measurement of the scram performance of the drives surrounding a drive, which exceeds the expected range of scram performance, will detect local variations and also provide assurance that local scram time limits are not exceeded. Continued monitoring of other drives exceeding the expected range of scram times provides surveillance of possible anomalous performance.

The test schedule provides reasonable assurance of detection of slow drives before system deterioration beyond the limits of Specification 3.3.C. The program was developed on the basis of the statistical approach outlined above and judgement. The occurrence of scram times within the limits, but significantly longer than average, should be viewed as an indication of a systematic problem with control rod drives, especially if the number of drives exhibiting such scram times exceeds eight, which is the allowable number of inoperable rods.

3/4.3.G Control Rod Scram Accumulators

The control rod scram accumulators are part of the control rod drive system and are provided to ensure that the control rods scram under varying reactor conditions. The control rod scram accumulators store sufficient energy to fully insert a control rod at any reactor vessel pressure. The accumulator is a hydraulic cylinder with a free floating piston. The piston separates the water used to scram the control rods from the nitrogen which provides the required energy. The scram accumulators are necessary to scram the control rods within the required insertion times.

Control rods with inoperable accumulators are declared inoperable and Specification 3.3.C then applies. This prevents a pattern of inoperable accumulators that would result in less reactivity insertion on a scram than has been analyzed even though control rods with inoperable accumulators may still be inserted with normal drive water pressure. OPERABILITY of the accumulator ensures that there is a means available to insert the control rods even under the most unfavorable depressurization of the reactor.

3/4.3.H Control Rod Drive Coupling

Control rod dropout accidents can lead to significant core damage. If coupling integrity is maintained, the possibility of a rod drop accident is eliminated. Neutron instrumentation response to rod movement may provide verification that a rod is following its drive. Absence of such response to drive movement may indicate an uncoupled condition, or may be due to the lack of proximity of the drive to the instrumentation. However, the overtravel position feature provides a positive check, as only uncoupled drives may reach this position.

3/4.3.I Control Rod Position Indication System (RPIS)

In order to ensure that the control rod patterns can be followed and therefore that other parameters are within their limits, the control rod position indication system must be OPERABLE. Normal control rod position is displayed by two-digit indication to the operator from position 00 to 48. Each even number is a latching position, whereas each odd number provides information while the rod is in-motion and inputs for rod drift annunciation. The ACTION statement provides for the condition where no positive information is displayed for a large portion or all of the rod's travel. Usually, only one digit of one or two of a rod's positions is unavailable with a faulty RPIS, and the control rod may be located in a known position. However, there are several alternate methods for determining control rod position including the full core display, the four rod display, the rod worth minimizer, and the process computer. Additionally, there are independent "full-in" and "full-out" indicators at the 00 and 48 positions. Another method to determine position would be to move the control rod, by single notch movement, to a position with an OPERABLE position indicator. The original position would then be established and the control rod could be returned to its original position by single notch movement. As long as no control rod drift alarms are received, the position of the control rod would then be known.

3/4.3.J Control Rod Drive Housing Support

The control rod housing support restricts the outward movement of a control rod to less than 3 inches in the extremely remote event of a housing failure. The amount of reactivity which could be added by this small amount of rod withdrawal, which is less than a normal single withdrawal increment, will not contribute to any damage to the primary coolant system. The design basis is given in Section 6.6.3 of the SAR. This support is not required if the reactor coolant system is at atmospheric pressure, since there would then be no driving force to rapidly eject a drive housing.

3/4.3.K Scram Discharge Volume Vent and Drain Valves

The scram discharge volume is required to be OPERABLE so that it will be available when needed to accept discharge water from the control rods during a reactor scram and will isolate the reactor coolant system from the containment when required. The operability of the scram discharge volume vent and drain valves assures the proper venting and draining of the volume, so that water accumulation in the volume does not occur. These specifications designate the minimum acceptable level of scram discharge volume vent and drain valve OPERABILITY, provide for the periodic verification that the valves are open, and for the testing of these valves under reactor scram conditions during each refueling outage.

3/4.3.L Rod Worth Minimizer

Control rod withdrawal and insertion sequences are established to assure that the maximum insequence individual control rod or control rod segments which are withdrawn at any time during the fuel cycle could not be worth enough to result in a peak fuel enthalpy greater than 280 cal/gm in the event of a control rod drop accident. These sequences are developed prior to initial operation of the unit following any refueling outage and the requirement that an operator follow these sequences is supervised by the RWM or a second technically qualified individual. These sequences are developed to limit reactivity worth of control rods and, together with the integral rod velocity limiters and the action of the control rod drive system, limit potential reactivity insertion such that the results of a control rod drop accident will not exceed a maximum fuel energy content of 280 cal/gm. The peak fuel enthalpy of 280 cal/gm is below the energy content at which rapid fuel dispersal and primary system damage have been found to occur based on experimental data. Therefore, the energy deposited during a postulated rod drop accident is significantly less than that required for rapid fuel dispersal.

The analysis of the control rod drop accident was originally presented in Sections 7.9.3, 14.2.1.2, and 14.2.1.4 of the SAR. Improvements in analytical capability have allowed a more refined analysis of the control rod drop accident which is discussed below.

Every operating cycle the peak fuel rod enthalpy rise is determined by comparing cycle specific parameters with the results of parametric analyses. This peak fuel rod enthalpy is then compared to the analysis limit of 280 cal/gm to demonstrate compliance for that operating cycle. If the cycle specific parameters are outside the range used in the parametric study, an extension of the enthalpy may be required. Some of the cycle specific parameters used in the analysis are: maximum control rod worth, Doppler coefficient, effective delayed neutron fraction and maximum four bundle local peaking factor. The NRC approved methodology listed in Specification 6.6.A.4 provides a detailed description of the methodology used in performing the rod drop analyses.

BASES

The rod worth minimizer provides automatic supervision to assure that out-of-sequence control rods will not be withdrawn or inserted, i.e., it limits operator deviations from planned withdrawal sequences (reference SAR Section 7.9). It serves as a backup to procedural control of control rod worth. In the event that the rod worth minimizer is out-of-service when required, a second licensed operator or other technically qualified individual who is present at the reactor console can manually fulfill the control rod pattern conformance function of the rod worth minimizer. In this case, the normal procedural controls are backed up by independent procedural controls to assure conformance.

3/4.3.M Rod Block Monitor

The rod block monitor (RBM) is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power operation. Two channels are provided, and one of these may be bypassed from the console for maintenance and/or testing. Tripping of one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage. This system backs up the operator, who withdraws control rods according to a written sequence. The specified restrictions with one channel out-of-service conservatively assure that fuel damage will not occur due to rod withdrawal errors when this condition exists.

3/4.3.N Economic Generation Control System

Operation of the facility with the economic generation control system (EGC) (automatic flow control) is limited to the range of 65% to 100% of rated core flow. In this flow range and above 20% of RATED THERMAL POWER, the reactor could safely tolerate a rate of change of load of 8 MWe/sec (reference SAR Section 7.3.6). Limits within the EGC and the flow control system prevent rates of change greater than approximately 4 MWe/sec. When EGC is in operation, this fact will be indicated on the main control room console.

3.3 - LIMITING CONDITIONS FOR OPERATION

A. SHUTDOWN MARGIN (SDM)

The SHUTDOWN MARGIN (SDM) shall be equal to or greater than:

1. 0.35% Δk/k with the highest worth control rod analytically determined, or

2. 0.25% Δk/k with the highest worth control rod determined by test.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, 3, 4, and 5.

ACTION:

With the SHUTDOWN MARGIN less than specified:

1. In OPERATIONAL MODE 1 or 2, restore the required SHUTDOWN MARGIN within 6 hours or be in at least HOT SHUTDOWN within the next 12 hours.

2. In OPERATIONAL MODE 3 or 4, immediately verify all insertable control rods to be fully inserted and suspend all activities that could reduce the SHUTDOWN MARGIN. In OPERATIONAL MODE 4, establish SECONDARY CONTAINMENT INTEGRITY within 8 hours.

3. In OPERATIONAL MODE 5, suspend CORE ALTERATION(s) and other activities that could reduce the SHUTDOWN MARGIN and fully insert all insertable control rods within 1 hour. Establish SECONDARY CONTAINMENT INTEGRITY within 8 hours.

4.3 - SURVEILLANCE REQUIREMENTS

A. SHUTDOWN MARGIN

The SHUTDOWN MARGIN shall be determined to be equal to or greater than that specified at any time during the operating cycle:

1. By demonstration, prior to or during the first startup after each refueling outage.

2. Within 24 hours after detection of a withdrawn control rod that is immovable, as a result of excessive friction or mechanical interference, or known to be unscrammable. The required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or unscrammable control rod.

3. By calculation, prior to each fuel movement during the fuel loading sequence.

3.3 - LIMITING CONDITIONS FOR OPERATION

B. Reactivity Anomalies

The reactivity equivalence of the difference between the actual ROD DENSITY and the predicted ROD DENSITY shall not exceed 1% Δk/k.

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2.

ACTION:

With the reactivity equivalence difference exceeding 1% Δk/k, within 12 hours perform an analysis to determine and explain the cause of the reactivity difference; operation may continue if the difference is explained and corrected.

With the provisions of the ACTION above not met, be in at least HOT SHUTDOWN within the next 12 hours.

4.3 - SURVEILLANCE REQUIREMENTS

B. Reactivity Anomalies

The reactivity equivalence of the difference between the actual ROD DENSITY and the predicted ROD DENSITY shall be verified to be less than or equal to 1% Δk/k:

1. During the first startup following CORE ALTERATION(s), and
2. At least once per 31 effective full power days.

3.3 - LIMITING CONDITIONS FOR OPERATION

C. Control Rod OPERABILITY

All control rods shall be OPERABLE.

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2.

ACTION:

1. With one control rod inoperable due to being immovable as a result of excessive friction or mechanical interference, or known to be unscrammable:
   a. Within one hour:
      1) Verify that the inoperable control rod, if withdrawn, is separated from all other inoperable withdrawn control rods by at least two control cells in all directions.
      2) Disarm the associated directional control valves[a] either:
         a) Electrically, or
         b) Hydraulically by closing the drive water and exhaust water isolation valves.
   b. With the provisions of ACTION 1.a above not met, be in at least HOT SHUTDOWN within the next 12 hours.

4.3 - SURVEILLANCE REQUIREMENTS

C. Control Rod OPERABILITY

1. When above the low power setpoint of the RWM, all withdrawn control rods not required to have their directional control valves disarmed electrically or hydraulically shall be demonstrated OPERABLE by moving each control rod at least one notch:
   a. At least once per 7 days, and
   b. At least once per 24 hours when any control rod is immovable as a result of excessive friction or mechanical interference, or known to be unscrammable.
2. All control rods shall be demonstrated OPERABLE by performance of Surveillance Requirements 4.3.D, 4.3.G, 4.3.H and 4.3.I.

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a May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.

3.3 - LIMITING CONDITIONS FOR OPERATION

4.3 - SURVEILLANCE REQUIREMENTS

c. Comply with Surveillance Requirement 4.3.A.2 within 24 hours or be in HOT SHUTDOWN within the next 12 hours.

2. With one or more control rods scrammable but inoperable for causes other than addressed in ACTION 3.3.C.1 above:

a. If the inoperable control rod(s) is withdrawn, within one hour:

1) Verify that the inoperable withdrawn control rod(s) is separated from all other inoperable withdrawn control rods by at least two control cells in all directions, and

2) Demonstrate the insertion capability of the inoperable withdrawn control rod(s) by inserting the inoperable withdrawn control rod(s) at least one notch by drive water pressure within the normal operating range.[b]

b. With the provisions of ACTION 2.a above not met, fully insert the inoperable withdrawn control rod(s) and disarm the associated directional control valves[a] either:

1) Electrically, or

2) Hydraulically by closing the drive water and exhaust water isolation valves.

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b The inoperable control rod may then be withdrawn to a position no further withdrawn than its position when found to be inoperable.

a May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.

3.3 - LIMITING CONDITIONS FOR OPERATION

c. If the inoperable control rod(s) is fully inserted, within one hour disarm the associated directional control valves[a] either:

1) Electrically, or

2) Hydraulically by closing the drive water and exhaust water isolation valves.

3. With the provisions of ACTION 2 above not met, be in at least HOT SHUTDOWN within the next 12 hours.

4. With more than 8 control rods inoperable, be in at least HOT SHUTDOWN within 12 hours.

4.3 - SURVEILLANCE REQUIREMENTS

---

a May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.

3.3 - LIMITING CONDITIONS FOR OPERATION

D. Maximum Scram Insertion Times

The maximum scram insertion time of each control rod from the fully withdrawn position to 90% insertion, based on de-energization of the scram pilot valve solenoids as time zero, shall not exceed 7 seconds.

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2.

ACTION:

With the maximum scram insertion time of one or more control rods exceeding 7 seconds:

1. Declare the control rod(s) exceeding the above maximum scram insertion time inoperable, and
2. When operation is continued with three or more control rods with maximum scram insertion times in excess of 7 seconds, perform Surveillance Requirement 4.3.D.3 at least once per 60 days of POWER OPERATION.

With the provisions of the ACTION(s) above not met, be in at least HOT SHUTDOWN within 12 hours.

4.3 - SURVEILLANCE REQUIREMENTS

D. Maximum Scram Insertion Times

The maximum scram insertion time of the control rods shall be demonstrated through measurement with reactor coolant pressure greater than 800 psig and, during single control rod scram time tests, with the control rod drive pumps isolated from the accumulators:

1. For all control rods prior to THERMAL POWER exceeding 40% of RATED THERMAL POWER:
   a. following CORE ALTERATION(s), or
   b. after a reactor shutdown that is greater than 120 days,
2. For specifically affected individual control rods[a] following maintenance on or modification to the control rod or control rod drive system which could affect the scram insertion time of those specific control rods, and
3. For at least 10% of the control rods, on a rotating basis, at least once per 120 days of POWER OPERATION.

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a The provisions of Specification 4.0.D are not applicable provided this surveillance is conducted prior to exceeding 40% of RATED THERMAL POWER.

3.3 - LIMITING CONDITIONS FOR OPERATION

E. Average Scram Insertion Times

The average scram insertion time of all OPERABLE control rods from the fully withdrawn position, based on de-energization of the scram pilot valve solenoids as time zero, shall not exceed any of the following:

% Inserted From Fully Withdrawn	Avg. Scram Insertion Times (sec)
5	0.375
20	0.900
50	2.00
90	3.50

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2.

ACTION:

With the average scram insertion time exceeding any of the above limits, be in at least HOT SHUTDOWN within 12 hours.

4.3 - SURVEILLANCE REQUIREMENTS

E. Average Scram Insertion Times

The control rod average scram times shall be demonstrated by scram time testing from the fully withdrawn position as required by Surveillance Requirement 4.3.D.

3.3 - LIMITING CONDITIONS FOR OPERATION

F. Group Scram Insertion Times

The average of the scram insertion times, from the fully withdrawn position, for the three fastest control rods of all groups of four control rods in a two-by-two array, based on de-energization of the scram pilot valve solenoids as time zero, shall not exceed any of the following:

% Inserted From Fully Withdrawn	Avg. Scram Insertion Times (sec)
5	0.398
20	0.954
50	2.120
90	3.800

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2.

ACTION:

With the average scram insertion times of control rods exceeding the above limits:

1. Declare the control rods exceeding the above average scram insertion times inoperable until an analysis is performed to determine that required scram reactivity remains for the slow four control rod group, and
2. When operation is continued with an average scram insertion time(s) in excess of the average scram insertion time limit, perform Surveillance Requirement 4.3.D.3 at least once per 60 days of POWER OPERATION.

With the provisions of the ACTION(s) above not met, be in at least HOT SHUTDOWN within 12 hours.

4.3 - SURVEILLANCE REQUIREMENTS

F. Group Scram Insertion Times

All control rods shall be demonstrated OPERABLE by scram time testing from the fully withdrawn position as required by Surveillance Requirement 4.3.D.

3.3 - LIMITING CONDITIONS FOR OPERATION

G. Control Rod Scram Accumulators

All control rod scram accumulators shall be OPERABLE.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 5[a].

ACTION:

In OPERATIONAL MODE 1 or 2:

a. With one control rod scram accumulator inoperable, within 8 hours:

1) Restore the inoperable accumulator to OPERABLE status, or

2) Declare the control rod associated with the inoperable accumulator inoperable.

b. With the provisions of ACTION 1.a above not met, be in at least HOT SHUTDOWN within the next 12 hours.

c. With more than one control rod scram accumulator inoperable, declare the associated control rods inoperable and:

4.3 - SURVEILLANCE REQUIREMENTS

G. Control Rod Scram Accumulators

Each control rod scram accumulator shall be determined OPERABLE at least once per 7 days by verifying that the indicated pressure is $\geq$800 psig unless the control rod is fully inserted and disarmed, or scrammed.

a In OPERATIONAL MODE 5, this Specification is applicable for the accumulators associated with each withdrawn control rod and is not applicable to control rods removed per Specification 3.10.D or 3.10.E.

3.3 - LIMITING CONDITIONS FOR OPERATION

1) If the control rod associated with any inoperable scram accumulator is withdrawn, immediately verify that at least one control rod drive pump is operating by inserting at least one withdrawn control rod at least one notch. With no control rod drive pump operating, immediately place the reactor mode switch in the Shutdown position.

2) Fully insert the inoperable control rods and disarm the associated directional control valves[b] either:

a) Electrically, or

b) Hydraulically by closing the drive water and exhaust water isolation valves.

d. With the provisions of ACTION 1.c.2 above not met, be in at least HOT SHUTDOWN within 12 hours.

2. In OPERATIONAL MODE 5[a]:

a. With one withdrawn control rod with its associated scram accumulator inoperable, fully insert the affected control rod and disarm the associated directional control valves[b] within one hour, either:

4.3 - SURVEILLANCE REQUIREMENTS

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a In OPERATIONAL MODE 5, this Specification is applicable for the accum[illegible]lators associated with each withdrawn control rod a[illegible] not applicable to control rods removed per Specification 3.10.D or 3.10.E.

b May be [illegible] intermittently, under administrative control, to permit testing associated with restoring the control roc [illegible] OPERABLE status.

3.3 - LIMITING CONDITIONS FOR OPERATION

4.3 - SURVEILLANCE REQUIREMENTS

1) Electrically, or

2) Hydraulically by closing the drive water and exhaust water isolation valves.

b. With more than one withdrawn control rod with the associated scram accumulator inoperable or no control rod drive pump operating, immediately place the reactor mode switch in the Shutdown position.

3.3 - LIMITING CONDITIONS FOR OPERATION

H. Control Rod Drive Coupling

All control rods shall be coupled to their drive mechanisms.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, and 5[a].

ACTION:

1. In OPERATIONAL MODE 1 or 2 with one control rod not coupled to its associated drive mechanism, within 2 hours:

    a. If permitted by the RWM, insert the control rod drive mechanism to accomplish recoupling and verify recoupling by withdrawing the control rod, and:

        1) Observing any indicated response of the nuclear instrumentation, and

        2) Demonstrating that the control rod will not go to the overtravel position.

    b. If not permitted by the RWM or, if recoupling is not accomplished in accordance with ACTION 1.a above, then declare the control rod inoperable, fully insert the control rod and disarm the associated directional control valves[b] either:

        1) Electrically, or

4.3 - SURVEILLANCE REQUIREMENTS

H. Control Rod Drive Coupling

Each affected control rod shall be demonstrated to be coupled to its drive mechanism by observing any indicated response of the nuclear instrumentation while withdrawing the control rod to the fully withdrawn position and then verifying that the control rod drive does not go to the overtravel position:

1. Prior to re[illegible]iticality after completing [illegible] ALTERATION(s) that could have affected the control rod drive coupling integrity,

2. Anytime the control rod is withdrawn to the "Full out" position in subsequent operation, and

3. Following maintenance on or modification to the control rod or control rod drive system which could have affected the control rod drive coupling integrity.

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a In OPERATIONAL MODE 5, this Specification is applicable for withdrawn control rods and is not applicable to control rods removed per Specification 3.10.D or 3.10.E.

b May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.

3.3 - LIMITING CONDITIONS FOR OPERATION

4.3 - SURVEILLANCE REQUIREMENTS

2) Hydraulically by closing the drive water and exhaust water isolation valves.

2. With the provisions of ACTION 1 above not met, be in at least HOT SHUTDOWN within 12 hours.

3. In OPERATIONAL MODE 5[a] with a withdrawn control rod not coupled to its associated drive mechanism, within 2 hours:

   a. Insert the control rod to accomplish recoupling and verify recoupling by withdrawing control rod and demonstrating that the control rod will not go to the overtravel position, or

   b. If recoupling is not accomplished, declare the control rod inoperable, fully insert the control rod and disarm the associated directional control valves[b] within one hour, either:

      1) Electrically, or

      2) Hydraulically by closing the drive water and exhaust water isolation valves.

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a In OPERATIONAL MODE 5, this Specification is applicable for withdrawn control rods and is not applicable to control rods removed per Specification 3.10.D or 3.10.E.

b May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.

3.3 - LIMITING CONDITIONS FOR OPERATION

I. Control Rod Position Indication System

All control rod position indicators shall be OPERABLE.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, and 5[a].

ACTION:

1. In OPERATIONAL MODE 1 or 2 with one or more control rod position indicators inoperable, within one hour either:
   a. Determine the position of the control rod by an alternate method, or
   b. Move the control rod to a position with an OPERABLE position indicator, or
   c. Declare the control rod inoperable, fully insert the inoperable withdrawn control rod(s), and disarm the associated directional control valves[b] either:
      1) Electrically, or
      2) Hydraulically by closing the drive water and exhaust water isolation valves.

4.3 - SURVEILLANCE REQUIREMENTS

I. Control Rod Position Indication System

The control rod position indication system shall be determined OPERABLE by verifying:

1. At least once per 24 hours that the position of each control rod is indicated.
2. That the indicated control rod position changes during the movement of the control rod drive when performing Surveillance Requirement 4.3.C.1.
3. That the control rod position indicator corresponds to the control rod position indicated by the "Full out" position indicator when performing Surveillance Requirement 4.3.H.2.

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a In OPERATIONAL MODE 5, this Specification is applicable for withdrawn control rods and is not applicable to control rods removed per Specification 3.10.D or 3.10.E.

b May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod(s) to OPERABLE status.

3.3 - LIMITING CONDITIONS FOR OPERATION

2. With the provisions of ACTION 1 above not met, be in at least HOT SHUTDOWN within the next 12 hours.

3. In OPERATIONAL MODE 5[a] with a withdrawn control rod position indicator inoperable:

   a. Mcve the control rod to a position with an OPERABLE position indicator, or

   b. Fully insert the control rod.

4.3 - SURVEILLANCE REQUIREMENTS

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a In OPERATIONAL MODE 5, this Specification is applicable for withdrawn control rods and is not applicable to control rods removed per Specification 3.10.D or 3.10.E.

3.3 - LIMITING CONDITIONS FOR OPERATION

J. Control Rod Drive Housing Support

The control rod drive housing support shall be in place.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, and 3.

ACTION:

With the control rod drive housing support not in place, be in at least HOT SHUTDOWN within 12 hours and in at least COLD SHUTDOWN within the following 24 hours.

4.3 - SURVEILLANCE REQUIREMENTS

J. Control Rod Drive Housing Support

The control rod drive housing support shall be verified to be in place by a visual inspection prior to startup any time it has been disassembled or when maintenance has been performed in the control rod drive housing support area.

3.3 - LIMITING CONDITIONS FOR OPERATION

K. SDV Vent and Drain Valves

All scram discharge volume (SDV) vent and drain valves shall be OPERABLE.

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2.

ACTION:

1. With one scram discharge volume vent valve and/or one scram discharge volume drain valve inoperable and open, restore the inoperable valve(s) to OPERABLE status within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours.
2. With any scram discharge volume vent valve(s) and/or any scram discharge volume drain valve(s) otherwise inoperable, restore at least one vent valve and one drain valve to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours.

4.3 - SURVEILLANCE REQUIREMENTS

K. SDV Vent and Drain Valves

The scram discharge volume vent and drain valves shall be demonstrated OPERABLE:

1. At least once per 31 days by verifying each valve to be open[a], and
2. At least once per 92 days by cycling each valve through at least one complete cycle of travel.
3. At least once per 18 months, the scram discharge volume vent and drain valves shall be demonstrated to:
   a. Close within 30 seconds after receipt of a signal for control rods to scram, and
   b. Open after the scram signal is reset.

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a These valves may be closed intermittently for testing under administrative controls.

3.3 - LIMITING CONDITIONS FOR OPERATION

L. Rod Worth Minimizer (RWM)

The rod worth minimizer (RWM) shall be OPERABLE.

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2[a], when THERMAL POWER is less than or equal to 10% of RATED THERMAL POWER.

ACTION:

With the RWM inoperable, verify control rod movement and compliance with the prescribed control rod pattern by a second licensed operator or technically qualified individual who is present at the reactor control console. Otherwise, control rod movement may be made only by actuating the manual scram or placing the reactor mode switch in the Shutdown position.

4.3 - SURVEILLANCE REQUIREMENTS

L. Rod Worth Minimizer (RWM)

The RWM shall be demonstrated OPERABLE:

1. By verifying that the control rod patterns and sequence input to the RWM computer are correctly loaded following any loading of the program into the computer.

2. In OPERATIONAL MODE 2 within 8 hours prior to withdrawal of control rods for the purpose of making the reactor critical:

   a. by verifying proper indication of the selection error of at least one out-of-sequence control rod.

   b. by verifying the rod block function by demonstrating inability to withdraw an out-of-sequence control rod.

3. In OPERATIONAL MODE 1 prior to reducing THERMAL POWER below 10% of RATED THERMAL POWER:

   a. by verifying proper indication of the selection error of at least one out-of-sequence control rod.

   b. by verifying the rod block function by demonstrating inability to withdraw an out-of-sequence control rod.

---

a Entry into OPERATIONAL MODE 2 and withdrawal of selected control rods is permitted for the purpose of determining the OPERABILITY of the RWM prior to withdrawal of control rods for the purpose of bringing the reactor to criticality.

3.3 - LIMITING CONDITIONS FOR OPERATION

M. Rod Block Monitor (RBM)

Both rod block monitor (RBM) CHANNEL(s) shall be OPERABLE.

APPLICABILITY:

OPERATIONAL MODE 1, when thermal power is greater than or equal to 30% of RATED THERMAL POWER.

ACTION:

1. With one RBM CHANNEL inoperable:

   a. Verify that the reactor is not operating in a LIMITING CONTROL ROD PATTERN, and

   b. Restore the inoperable RBM CHANNEL to OPERABLE status within 24 hours.

2. With the provisions of ACTION 1 above not met, place the inoperable rod block monitor CHANNEL in the tripped condition within the next one hour.

3. With both RBM CHANNEL(s) inoperable, place at least one inoperable rod block monitor CHANNEL in the tripped condition within one hour.

4.3 - SURVEILLANCE REQUIREMENTS

M. Rod Block Monitor (RBM)

Each of the required RBM CHANNEL(s) shall be demonstrated OPERABLE by performance of a:

1. CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION at the frequencies and for the OPERATIONAL MODE(s) specified in Table 4.2.E-1.

2. CHANNEL FUNCTIONAL TEST prior to control rod withdrawal when the reactor is operating in a LIMITING CONTROL ROD PATTERN, but no more often than daily.

3.3 - LIMITING CONDITIONS FOR OPERATION

N. Economic Generation Control (EGC) System

The economic generation control (EGC) system may be in operation with automatic flow control provided:

a. Core flow is within 65% to 100% of rated core flow, and

b. THERMAL POWER is ≥20% of RATED THERMAL POWER.

APPLICABILITY

OPERATIONAL MODE 1.

ACTION:

With core flow less than 65% or greater than 100% of rated core flow, or THERMAL POWER less than 20% of RATED THERMAL POWER, restore operation to within the limits within one hour. Otherwise, immediately remove the plant from EGC operation.

4.3 - SURVEILLANCE REQUIREMENTS

N. Economic Generation Control (EGC) System

The economic generation control system shall be demonstrated OPERABLE by verifying that core flow is within 65% to 100% of rated core flow and THERMAL POWER is ≥20% of RATED THERMAL POWER:

a. Prior to entry into EGC operation, and

b. At least once per 12 hours while operating in EGC.

3/4.3.A SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

The SHUTDOWN MARGIN limitation is a restriction to be applied principally to a new refueling pattern. Satisfaction of the limitation must be determined at the time of loading and must be such that it will apply to the entire subsequent fuel cycle. This determination is provided by core design calculations and administrative control of fuel loading patterns. These procedures include restrictions to allow only those intermediate fuel assembly configurations that have been shown to provide the required SHUTDOWN MARGIN.

Since core reactivity values will vary through core life as a function of fuel depletion and poison burnup, the demonstration of SHUTDOWN MARGIN will be performed during xenon free conditions and adjusted to 68°F to accommodate the current moderator temperature. The generalized form is that the reactivity of the core loading will be limited so the core can be made subcritical by at least R + 0.35% Δk/k or R + 0.25% Δk/k, as appropriate, with the strongest control rod fully withdrawn and all others fully inserted. Two different values are supplied in the Limiting Condition for Operation to provide for the different methods of determination of the highest control rod worth, either analytically or by test. This is due to the reduced uncertainty in the SHUTDOWN MARGIN test when the highest worth control rod is determined by demonstration. When SHUTDOWN MARGIN is determined by calculations not associated with a test, additional margin must be added to the specified SHUTDOWN MARGIN limit to account for uncertainties in the calculation.

The value of R in units of % Δk/k is the difference between the calculated beginning-of-life core reactivity, at the beginning of the operating cycle, and the calculated value of maximum core reactivity at any time later in the operating cycle, where it would be greater than at the beginning. The value of R shall include the potential SHUTDOWN MARGIN loss assuming full B_4C settling in all inverted poison tubes present in the core. R must be a positive quantity or zero and a new value of R must be determined for each new fuel cycle.

The value of % Δk/k in the above expression is provided as a finite, demonstrable, subcriticality margin. This margin may be demonstrated by full withdrawal of the strongest rod and partial withdrawal of an adjacent rod to a position calculated to add at least R + 0.25% Δk/k (or 0.35% Δk/k) in reactivity. Observation of subcriticality in this condition assures subcriticality with not only the strongest rod fully withdrawn but with a substantial margin beyond this condition. This reactivity characteristic has been a basic assumption in the analysis of plant performance and can be best demonstrated at the time of fuel loading, but the margin must also be determined anytime a control rod is incapable of insertion following a scram signal. Any control rod that is immovable as a result of excessive friction or mechanical interference, or is known to be unscrammable, per Specification 3.3.C, is considered to be incapable of insertion following a scram signal. It is important to note that a control rod can be electrically immovable, but scrammable, and no increase in SHUTDOWN MARGIN is required for these control rods.

3/4.3.B Reactivity Anomalies

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

3/4.3.C Control Rod OPERABILITY

Control rods are the primary reactivity control system for the reactor. In conjunction with the Reactor Protection System, the control rods provide the means for reliable control of reactivity changes to ensure the specified acceptable fuel design limits are not exceeded. This specification, along with others, assures that the performance of the control rods in the event of an accident or transient, meets the assumptions used in the safety analysis. Of primary concern is the trippability of the control rods. Other causes for inoperability are addressed in other Specifications following this one. However, the inability to move a control rod which remains trippable does not prevent the performance of the control rod's safety function.

The specification requires that a rod be taken out-of-service if it cannot be moved with drive pressure. Damage within the control rod drive mechanism could be a generic problem, therefore with a control rod immovable because of excessive friction or mechanical interference, operation of the reactor is limited to a time period which is reasonable to determine the cause of the inoperability and at the same time prevent operation with a large number of inoperable control rods. Control rods that are inoperable due to exceeding allowed scram times, but are movable by control rod drive pressure, need not be disarmed electrically if the shutdown margin provisions are met for each position of the affected rod(s).

If the rod is fully inserted and then disarmed electrically or hydraulically, it is in a safe position of maximum contribution to shutdown reactivity. (Note: To disarm the drive electrically, four amphenol-type plug connectors are removed from the drive insert and withdrawal solenoids, rendering the drive immovable. This procedure is equivalent to valving out the drive and is preferred, as drive water cools and minimizes crud accumulation in the drive.). If it is disarmed

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electrically in a non-fully inserted position, that position shall be consistent with the SHUTDOWN MARGIN limitation stated in Specification 3.3.A. This assures that the core can be shut down at all times with the remaining control rods, assuming the strongest OPERABLE control rod does not insert. The occurrence of more than eight inoperable control rods could be indicative of a generic control rod drive problem which requires prompt investigation and resolution.

In order to reduce the potential for Control Rod Drive (CRD) damage and more specifically, collet housing failure, a program of disassembly and inspection of CRDs is conducted during or after each refueling outage. This program follows the recommendations of General Electric SIL-139 with nondestructive examination results compiled and reported to General Electric on collet housing cracking problems.

The required surveillance intervals are adequate to determine that the rods are OPERABLE and not so frequent as to cause excessive wear on the system components.

3/4.3.D Control Rod Maximum Scram Insertion Times;

3/4.3.E Control Rod Average Scram Insertion Times; and

3/4.3.F Four Control Rod Group Scram Insertion Times

These specifications ensure that the control rod insertion times are consistent with those used in the safety analyses. The control rod system is analyzed to bring the reactor subcritical at a rate fast enough to prevent fuel damage, i.e., to prevent the MCPR from becoming less than the fuel cladding integrity Safety Limit. The analyses demonstrate that if the reactor is operated within the limitation set in Specification 3.11.C, the negative reactivity insertion rates associated with the scram performance (as adjusted for statistical variation in the observed data) result in protection of the MCPR Safety Limit.

Analysis of the limiting power transient shows that the negative reactivity rates, resulting from the scram with the average response of all the drives, as given in the above specification, provide the required protection, and MCPR remains greater than the fuel cladding integrity SAFETY LIMIT. In the analytical treatment of most transients, 290 milliseconds are allowed between a neutron sensor reaching the scram point and the start of motion of the control rods. This is adequate and conservative when compared to the typically observed time delay of about 210 milliseconds. Approximately 90 milliseconds after neutron flux reaches the trip point, the pilot scram valve solenoid de-energizes and 120 milliseconds later the control rod motion is estimated to actually begin. However, 200 milliseconds rather than 120 milliseconds is conservatively assumed for this time interval in the transient analyses and is also included in the allowable scram insertion times specified in Specifications 3.3.D, 3.3.E, and 3.3.F. In the statistical treatment of the limiting transients, a statistical distribution of total scram delay is used rather than the bounding value described above.

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The performance of the individual control rod drives is monitored to assure that scram performance is not degraded. Observed plant data or Technical Specification limits were used to determine the average scram performance used in the transient analyses, and the results of each set of control rod scram tests performed during the current cycle are compared against earlier results to verify that the performance of the control rod insertion system has not changed significantly.

If test results should be determined to fall outside of the statistical population defining the scram performance characteristics used in the transient analyses, a re-determination of thermal margin requirements is undertaken as required by Specification 3.11.C. A smaller test sample than that required by these specifications is not statistically significant and should not be used in the re-determination of thermal margins. Individual control rod drives with excessive scram times can be fully inserted into the core and de-energized in the manner of an inoperable rod drive provided the allowable number of inoperable control rod drives is not exceeded. In this case, the scram speed of the drive shall not be used as a basis in the re-determination of thermal margin requirements. For excessive average scram insertion times, only the individual control rods in the two-by-two array which exceed the allowed average scram insertion time are considered inoperable.

The scram times for all control rods are measured at the time of each refueling outage. Experience with the plant has shown that control drive insertion times vary little through the operating cycle; hence no re-assessment of thermal margin requirements is expected under normal conditions. The history of drive performance accumulated to date indicates that the 90% insertion times of new and overhauled drives approximate a normal distribution about the mean which tends to become skewed toward longer scram times as operating time is accumulated. The probability of a drive not exceeding the mean 90% insertion time by 0.75 seconds is greater than 0.999 for a normal distribution. The measurement of the scram performance of the drives surrounding a drive, which exceeds the expected range of scram performance, will detect local variations and also provide assurance that local scram time limits are not exceeded. Continued monitoring of other drives exceeding the expected range of scram times provides surveillance of possible anomalous performance.

The test schedule provides reasonable assurance of detection of slow drives before system deterioration beyond the limits of Specification 3.3.C. The program was developed on the basis of the statistical approach outlined above and judgement. The occurrence of scram times within the limits, but significantly longer than average, should be viewed as an indication of a systematic problem with control rod drives, especially if the number of drives exhibiting such scram times exceeds eight, which is the allowable number of inoperable rods.

<u>3/4.3.G</u> <u>Control Rod Scram Accumulators</u>

The control rod scram accumulators are part of the control rod drive system and are provided to ensure that the control rods scram under varying reactor conditions. The control rod scram accumulators store sufficient energy to fully insert a control rod at any reactor vessel pressure. The accumulator is a hydraulic cylinder with a free floating piston. The piston separates the water used to scram the control rods from the nitrogen which provides the required energy. The scram accumulators are necessary to scram the control rods within the required insertion times.

Control rods with inoperable accumulators are declared inoperable and Specification 3.3.C then applies. This prevents a pattern of inoperable accumulators that would result in less reactivity insertion on a scram than has been analyzed even though control rods with inoperable accumulators may still be inserted with normal drive water pressure. OPERABILITY of the accumulator ensures that there is a means available to insert the control rods even under the most unfavorable depressurization of the reactor.

3/4.3.H Control Rod Drive Coupling

Control rod dropout accidents can lead to significant core damage. If coupling integrity is maintained, the possibility of a rod drop accident is eliminated. Neutron instrumentation response to rod movement may provide verification that a rod is following its drive. Absence of such response to drive movement may indicate an uncoupled condition, or may be due to the lack of proximity of the drive to the instrumentation. However, the overtravel position feature provides a positive check, as only uncoupled drives may reach this position.

3/4.3.I Control Rod Position Indication System (RPIS)

In order to ensure that the control rod patterns can be followed and therefore that other parameters are within their limits, the control rod position indication system must be OPERABLE. Normal control rod position is displayed by two-digit indication to the operator from position 00 to 48. Each even number is a latching position, whereas each odd number provides information while the rod is in-motion and inputs for rod drift annunciation. The ACTION statement provides for the condition where no positive information is displayed for a large portion or all of the rod's travel. Usually, only one digit of one or two of a rod's positions is unavailable with a faulty RPIS, and the control rod may be located in a known position. However, there are several alternate methods for determining control rod position including the full core display, the four rod display, the rod worth minimizer, and the process computer. Additionally, there are independent "full-in" and "full-out" indicators at the 00 and 48 positions. Another method to determine position would be to move the control rod, by single notch movement, to a position with an OPERABLE position indicator. The original position would then be established and the control rod could be returned to its original position by single notch movement. As long as no control rod drift alarms are received, the position of the control rod would then be known.

3/4.3.J Control Rod Drive Housing Support

The control rod housing support restricts the outward movement of a control rod to less than 3 inches in the extremely remote event of a housing failure. The amount of reactivity which could be added by this small amount of rod withdrawal, which is less than a normal single withdrawal increment, will not contribute to any damage to the primary coolant system. The design basis is given in Section 6.6.3 of the SAR. This support is not required if the reactor coolant system is at atmospheric pressure, since there would then be no driving force to rapidly eject a drive housing.

3/4.3.K Scram Discharge Volume Vent and Drain Valves

The scram discharge volume is required to be OPERABLE so that it will be available when needed to accept discharge water from the control rods during a reactor scram and will isolate the reactor coolant system from the containment when required. The operability of the scram discharge volume vent and drain valves assures the proper venting and draining of the volume, so that water accumulation in the volume does not occur. These specifications designate the minimum acceptable level of scram discharge volume vent and drain valve OPERABILITY, provide for the periodic verification that the valves are open, and for the testing of these valves under reactor scram conditions during each refueling outage.

3/4.3.L Rod Worth Minimizer

Control rod withdrawal and insertion sequences are established to assure that the maximum insequence individual control rod or control rod segments which are withdrawn at any time during the fuel cycle could not be worth enough to result in a peak fuel enthalpy greater than 280 cal/gm in the event of a control rod drop accident. These sequences are developed prior to initial operation of the unit following any refueling outage and the requirement that an operator follow these sequences is supervised by the RWM or a second technically qualified individual. These sequences are developed to limit reactivity worth of control rods and, together with the integral rod velocity limiters and the action of the control rod drive system, limit potential reactivity insertion such that the results of a control rod drop accident will not exceed a maximum fuel energy content of 280 cal/gm. The peak fuel enthalpy of 280 cal/gm is below the energy content at which rapid fuel dispersal and primary system damage have been found to occur based on experimental data. Therefore, the energy deposited during a postulated rod drop accident is significantly less than that required for rapid fuel dispersal.

The analysis of the control rod drop accident was originally presented in Sections 7.9.3, 14.2.1.2, and 14.2.1.4 of the SAR. Improvements in analytical capability have allowed a more refined analysis of the control rod drop accident which is discussed below.

Every operating cycle the peak fuel rod enthalpy rise is determined by comparing cycle specific parameters with the results of parametric analyses. This peak fuel rod enthalpy is then compared to the analysis limit of 280 cal/gm to demonstrate compliance for that operating cycle. If the cycle specific parameters are outside the range used in the parametric study, an extension of the enthalpy may be required. Some of the cycle specific parameters used in the analysis are: maximum control rod worth, Doppler coefficient, effective delayed neutron fraction and maximum four bundle local peaking factor. The NRC approved methodology listed in Specification 6.6.A.4 provides a detailed description of the methodology used in performing the rod drop analyses.

The rod worth minimizer provides automatic supervision to assure that out-of-sequence control rods will not be withdrawn or inserted, i.e., it limits operator deviations from planned withdrawal sequences (reference SAR Section 7.9). It serves as a backup to procedural control of control rod worth. In the event that the rod worth minimizer is out-of-service when required, a second licensed operator or other technically qualified individual who is present at the reactor console can manually fulfill the control rod pattern conformance function of the rod worth minimizer. In this case, the normal procedural controls are backed up by independent procedural controls to assure conformance.

3/4.3.M Rod Block Monitor

The rod block monitor (RBM) is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power operation. Two channels are provided, and one of these may be bypassed from the console for maintenance and/or testing. Tripping of one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage. This system backs up the operator, who withdraws control rods according to a written sequence. The specified restrictions with one channel out-of-service conservatively assure that fuel damage will not occur due to rod withdrawal errors when this condition exists.

3/4.3.N Economic Generation Control System

Operation of the facility with the economic generation control system (EGC) (automatic flow control) is limited to the range of 65% to 100% of rated core flow. In this flow range and above 20% of RATED THERMAL POWER, the reactor could safely tolerate a rate of change of load of 8 MWe/sec (reference SAR Section 7.3.6). Limits within the EGC and the flow control system prevent rates of change greater than approximately 4 MWe/sec. When EGC is in operation, this fact will be indicated on the main control room console.

ATTACHMENT 4

EXISTING TECHNICAL SPECIFICATIONS

Technical Specification 3/4.3

"REACTIVITY CONTROL"

ATTACHMENT 4

DELETION OF CURRENT TECHNICAL SPECIFICATIONS

This technical specification amendment will replace the current section 3.3/4.3, Reactivity Control, for the Dresden Unit 2 and Unit 3 Technical Specifications. The specifications are replaced in their entirety with revised pages that combine the Unit 2 and Unit 3 specifications.

Delete the following pages:

DPR - 19	DPR - 25
3/4.3-1	3/4.3-1
3/4.3-2	3/4.3-2
3/4.3-3	3/4.3-3
3/4.3-4	3/4.3-4
3/4.3-5	3/4.3-5
3/4.3-6	B 3/4.3-6
3/4.3-7	B 3/4.3-7
3/4.3-8	B 3/4.3-8
3/4.3-9	B 3/4.3-9
3/4.3-10	B 3/4.3-10
3/4.3-11	B 3/4.3-11
3/4.3-12	B 3/4.3-12
3/4.3-13	B 3/4.3-13
B 3/4.3-14	B 3/4.3-14
B 3/4.3-15	B 3/4.3-15
B 3/4.3-16	B 3/4.3-16
B 3/4.3-17	B 3/4.3-17
B 3/4.3-18	B 3/4.3-18
B 3/4.3-19	B 3/4.3-19
B 3/4.3-20	B 3/4.3-20
B 3/4.3-21	B 3/4.3-21
--	B 3/4.3-22

ATTACHMENT 4

DELETION OF CURRENT TECHNICAL SPECIFICATIONS

This technical specification amendment will replace the current sections 3/4.3, Reactivity Control, for the Quad Cities Unit 1 and Unit 2 Technical Specifications. The specifications are replaced in there entirety with revised pages that combine the Unit 1 and Unit 2 specifications.

Delete the following pages:

DPR - 29	DPR - 30
3.3/4.3-1	3.3/4.3-1
3.3/4.3-2	3.3/4.3-1a
3.3/4.3-3	3.3/4.3-2
3.3/4.3-4	3.3/4.3-3
3.3/4.3-5	3.3/4.3-4
3.3/4.3-6	3.3/4.3-5
3.3/4.3-7	3.3/4.3-6
3.3/4.3-8	3.3/4.3-7
3.3/4.3-9	3.3/4.3-7a
3.3/4.3-10	3.3/4.3-8
3.3/4.3-11	3.3/4.3-9
3.3/4.3-12	3.3/4.3-10
3.3/4.3-13	3.3/4.3-11
3.3/4.3-14	
3.3/4.3-15	
3.3/4.3-16	
3.3/4.3-17	

ATTACHMENT 5

DRESDEN 2/3 DIFFERENCES

Technical Specification 3/4.3

"REACTIVITY CONTROL"

ATTACHMENT 5

COMPARISON OF DRESDEN UNIT 2 AND UNIT 3 TECHNICAL SPECIFICATIONS
FOR THE
IDENTIFICATION OF TECHNICAL DIFFERENCES

SECTION 3/4.3
"REACTIVITY CONTROL"

Commonwealth Edison has conducted a comparison review of the 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.) or punctuation but rather to identify areas which the Technical Specifications are technically different.

The review of the Section 3/4.3, "Reactivity Control" did not identify any technical differences in the LCO or SR sections. However, five differences were noted within the BASES of the Technical Specifications.

Unit 2:
Page B 3/4.3-14....."For the first fuel cycle, R was calculated to be not greater than 0.10% delta k."
Unit 3:
Page B 3/4.3-14....."The value of R shall include the potential shutdown margin loss assuming full B,C setting in all inverted poison tubes present in the core."
Resolution:
The proposed Bases delete the reference to the first fuel cycle and appropriately discuss the present method of handling the Boron settling of the inverted poison tubes for both units.

Unit 2:
Page B 3/4.3-14.....No Footnote.
Unit 3:
Page B 3/4.3-14.....Footnote......"NOTE: This change issued by letter dated 08/27/75 which noted that 0.02% delta k should be included in the value.
Resolution:
The value for boron settling is controlled within station procedures and therefore only a discussion of the problem is included in the Bases.

Unit 2:
Page B 3/4.3-19....."as specified in Specification 3.3.C"
Unit 3:
Page B 3/4.3-19....."observed scram performance (as adjusted for statistical variation in the observed data) result in protection of the MCPR safety limit."
Resolution:
The proposed Bases include the Unit 3 version of this information for both units.

Unit 2:
Page B 3/4.3-20....."The bounding value described above was used in the transient analysis."
Unit 3:
Page B 3/4.3-20....."In the statistical treatment of the limiting transients, a statistical distribution of total scram delay is used rather than the bounding value described above."
Resolution:

The proposed Bases reflect the current provision of adjusting the MCPR limit based on the results of the SCRAM times for both units.

Unit 2:
Page B 3/4.3-20....."vary little through the operating cycle."
Unit 3:
Page B 3/4.3-20....."vary little through the operating cycle; hence no reassessment of thermal margin requirements is expected under normal conditions."
Resolution:

The proposed Bases reflect the information gained through experience of SCRAM timing for both units.

ATTACHMENT 5

QUAD CITIES 1/2 DIFFERENCES

Technical Specification 3/4.3

"REACTIVITY CONTROL"

ATTACHMENT 5

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

SECTION 3/4.3
"REACTIVITY CONTROL"

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.) or punctuation but rather to identify areas which the Technical Specifications are technically different.

The review of Section 3/4.3, "Reactivity Control" did not reveal any technical differences.

ATTACHMENT 6

SIGNIFICANT HAZARDS CONSIDERATIONS AND ENVIRONMENTAL ASSESSMENT EVALUATION

Technical Specification 3/4.3

"<u>REACTIVITY CONTROL</u>"

ATTACHMENT 6

EVALUATION OF SIGNIFICANT HAZARDS CONSIDERATION

Commonwealth Edison has evaluated this proposed amendment and determined that it involves no significant haz rds 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.

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 the requirements for shutdown margin technical specifications implement proven STS guidelines that are applicable to Dresden and Quad Cities. Proposed Applicability of operational modes 1, 2, 3, 4 and 5 ensures that shutdown margin limitations are enforced when fuel is in the vessel and a potential exists for reactivity excursions. STS actions are added to avoid using the provisions of proposed Specification 3.0.C and to implement proven action allowances. Since STS provisions are implemented, the changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

c) Proposed changes for Specifications for Reactivity Anomalies, 3/4.3.B, include using the LCO, Applicability, action and Surveillance Requirements from the STS. Reactivity anomalies of concern are while operating in operational modes 1 or 2 and if the limit of 1% delta k/k is exceeded. Present and proposed provisions retain these operational modes and reactivity difference limitation. Proposed actions are taken from STS guidelines and allow 12 hours to perform an analysis to determine and explain the cause of the reactivity difference or the plant must be in at least hot shutdown within 12 hours. The 12 hours is a reasonable time frame to evaluate core conditions before requiring plant shutdown steps. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

d) Proposed Specification 3/4.3.C on Control Rod operability utilizes STS guidelines. These requirements are used to form acceptable requirements for Dresden and Quad Cities. The present reference to collet housing failures is deleted since this determination is made during CRD disassembly and inspection while the plant is shutdown and is not determined during plant operation. Proposed operability is in operational modes 1 and 2 when control rod insertion rates are considered in the accident analysis. Proposed actions for Control Rod operability implement proven STS guidelines that are applicable at Dresden and Quad Cities. The proposed Surveillance Requirements are based on STS provisions that will provide necessary demonstration of control rod operability. Since the present level of control rod operability is maintained, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

e) Proposed specifications for the Control Rod Maximum Scram Insertion Times, 3/4.3.D, add STS Applicability and action provisions including the 7 second maximum insertion time. Proposed Applicability of operational modes 1 and 2 covers the conditions where scram insertion times must be met for transient analysis considerations. Proposed actions from STS guidelines are consistent with current requirement to ensure that proper control rod operability is maintained when operation is continued with three or more control rods with maximum scram insertion times exceeding 7 seconds. With the maximum scram insertion time of one or more control rods exceeding 7 seconds, the control rods are declared inoperable and with three or more control rods exceeding the limit, scram insertion time tests are performed on an accelerated basis of at least once per 60 days. Proposed Surveillance Requirements for control rod scram time testing follows STS guidelines. The STS guidelines for control rod scram time testing have been utilized at other operating facilities with control rod drive systems similar to that at Dresden and Quad Cities and will maintain necessary assurances of control rod operability. The proposed frequency change for scram time demonstration is being evaluated for the impact on the operating limit Minimum Critical Power Ratio (MCPR) reported in the Core Operating Limits Report. The results of the analysis will be used to make any necessary adjustments to the operating limit MCPR to maintain the same degree of margin to the MCPR Safety Limit. This analysis will be completed prior to the approval of this amendment. The proposed changes maintain necessary operability of the Control Rod Maximum Scram Insertion Time criteria, and; therefore, do not involve a significant increase in the probability or consequences of an accident previously evaluated.

f) Proposed changes to the Specifications for the Control Rod Average Scram Insertion Times, 3/4.3.E, involve the retention of present requirements which are identical to the requirements set forth in the STS. Present Control Rod Average Scram Insertion Times are retained in operational modes 1 and 2 using STS guidelines. operational modes 1 and 2 cover the reactor operational conditions where control rod scram times are required by analysis. The proposed action allows 12 hours to reach hot shutdown and implements present intent to shutdown the reactor. Proposed Surveillance Requirements reference the SRs of 4.3.D which are discussed above. The proposed changes maintain at least the present level of operability and as such do not involve a significant increase in the probability or consequences of an accident previously evaluated.

g) Proposed changes to the Specifications for the Four Control Rod Group Scram Insertion Times, 3/4.3.F, involve retention of present requirements in accordance with the STS guidelines. Present Four Control Rod Group Scram Insertion Times are retained in operational modes 1 and 2 using STS guidelines. operational modes 1 and 2 cover the reactor operational conditions where control rod scram times are required by analysis. The proposed actions are based on STS guidelines and requires the slow control rods to be declared inoperable until an analysis is performed to determine that required scram reactivity remains for the slow four control rod group. If continued operation is allowed, accelerated scram time testing is imposed on a frequency of at least once per 60 days. Proposed Surveillance Requirements reference the SRs of 4.3.D which are discussed above. The proposed changes maintain at least the present level of operability and as such do not involve a significant increase in the probability or consequences of an accident previously evaluated.

h) Proposed Specifications for the Control Rod Scram Accumulators, 3/4.3.G, are based on STS guidelines. The proposed changes require all Control Rod Scram Accumulators to be operable in operational modes 1, 2, and 5 with applicability in operational mode 5 being only for withdrawn control rods. The proposed applicability covers all reactor operational modes where control rods can be withdrawn and insertion using accumulators may be required. The proposed actions implement STS guidelines and separately address operational modes 1 and 2 and operational mode 5. The proposed actions require the plant to be shutdown while in operational modes 1 or 2 with remedial steps not met or requires the reactor mode switch to be placed in shutdown if remedial steps are not met in operational mode 5. Present Surveillance Requirements are replaced with STS guidelines to check the pressure at least once per week of each control rod scram accumulator. Operability of the Control Rod Scram Accumulators is maintained at least at the present level of operability by the proposed changes, and as such, the changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

i) The proposed rewrite of the Control Rod Drive Coupling specifications utilizes STS guidelines in a manner that improves usability. The proposed Applicability of operational modes 1, 2 and 5 (for withdrawn control rods) addresses all reactor modes of operation where control rod drive coupling is necessary. Action provisions from STS guidelines are added to allow an attempt to recouple the control rod. STS requirements to insert and disarm the control rod are equivalent to present requirements. The intent of present Surveillance Requirements are maintained with the rewrite of proposed SR 4.3.H. The present level of operability for Control Rod Drive Coupling is maintained by the proposed changes. Since necessary control rod coupling requirements are maintained in the proposed changes, there is no significant increase in the probability or consequences of an accident previously evaluated.

j) Proposed Specification 3/4.3.I for the Control Rod Position Indication System is based on STS provisions. Operability of the Control Rod Position Indication System is required when control rod movement is allowed in operational modes 1, 2, and 5. The present Specifications (Quad Cities) do not contain action provisions that separately address the needs of the Control Rod Position Indication System in each of the operational modes in which the system is required operable. Proposed actions from the STS are used to address the different requirements for operational modes 1 and 2 versus operational mode 5. The present specifications (Quad Cities) require control rods that have an inoperable position indication to be moved to a known position or be inserted, scrammed and considered inoperable. The insertion and disarming of the control rod is sufficient without the additional requirement to scram the control rod, and as such, this requirement to also scram the control rod is deleted. Surveillance Requirement 4.3.I.1 to verify position indication of each control rod is required once per 24 hours. SRs are added to verify position indication during weekly notching of control rods. The proposed changes to the Control Rod Position Indication System specifications improve present requirements and, as such, do not involve a significant increase in the probability or consequences of an accident previously evaluated.

k) The proposed Specification for the Control Rod Drive Housing Support, 3/4.3.J, is based on STS provisions which are identical to the current requirements. Therefore, the proposed change is a change in format and not technical requirements. As a result, the format changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

l) Proposed Specification 3/4.3.K for the Scram Discharge Volume Vent and Drain Valves uses later operating BWR plants' specifications to develop LCO, Applicability and action requirements. All Scram Discharge Volume Vent and Drain Valves are required operable in operational modes 1 and 2 in order to provide a means to isolate the Scram Discharge Volume when needed for reactor scram purposes involving multiple control rods. The proposed actions consider the design of two isolation valves on each drain and vent line by allowing 24 hours to return a valve to operable status if one of the vent or drain valves on a line is inoperable or one vent valve and one drain valve on a SDV is inoperable. Only 8 hours is allowed for continued operation if the SDV Vent and Drain Valves are inoperable for reasons other than those just described. The proposed changes improve the present technical specification requirements for the SDV Vent and Drain Valves by providing needed LCO, Applicability and action requirements while maintaining at least the present level of operability. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

m) The proposed Specification for the Rod Worth Minimizer, 3/4.3.L, is based on present provisions which are equivalent to STS guidelines except for the requirements for the Rod Sequence Control System. Neither Dresden or Quad Cities have Rod Sequence Control Systems and therefore those requirements are not applicable. The Applicability requirements come from present requirements in operational modes 1 and 2 below 20% (proposed 10% for Quad Cities) rated thermal power. Exceptions in the Applicability allows entry into operational mode 2 to allow determination of operability prior to withdrawing control rods for the purpose of bringing the reactor critical. The intent of present actions is retained with the addition of the STS guideline to require that without use of the second licensed operator or technically qualified individual, control rod movement is only allowed by initiating a reactor scram. The proposed actions provide adequate assurance that the control rods will be withdrawn in accordance with the prescribed patterns without the necessity of requiring 12 control rods to be fully withdrawn before a substitute can be used for the RWM. Present SRs [illegible] ned by using STS wording and guidelines. The proposed changes to [illegible] nical specifications maintains necessary operability requirements for [illegible] RWM when required to perform the design function below 20% (proposed 10% for Quad Cities) rated thermal power. Since the present level of operability for the RWM is retained, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

Improved methodologies in the Rod Drop Accident (RDA) analysis methods (e.g. BNL-NUREG 28109, "Thermal Hydraulic Effects on Center Rod Drop Accidents in a BWR, October 1980," have shown that when above 10 percent power no RDA can occur with the peak fuel enthalpy being greater than the RDA design limit of 280 cal/gm. The installed sensors for the RWM low power setpoint actuation are capable of providing actuation within the revised limits. The probability or consequences of an accident is not increased with the proposed lowering of the low power setpoint.

n) The proposed Specification for RBM, 3/4.3.M, is written utilizing STS guidelines. Both RBM channels are required operable in operational mode 1 above 30% rated thermal power in accordance with present provisions. STS guidelines are used to rewrite present action statements to address the conditions where either one RBM channel is inoperable or where both RBM channels are inoperable. Safe conditions are established by the proposed actions by inhibiting control rod movement if the remedial measures are not met. Surveillance Requirements for the RBM implem[illegible]t present channel functional test frequencies, which are identical to STS requirements, when operating on a limiting control rod pattern and reference Table 4.2.E-1 for regular surveillance testing. The proposed changes to the specifications for the RBM add to and improve the usability of the present requirements while maintaining at least the present level of operability. The current specifications require both RBMs to be operable when operating on a limiting control rod pattern. The definition limiting control rod pattern has been added to the proposed technical specifications in accordance with STS. The philosophy of a limiting control rod pattern used currently is different than the definition in the STS. The current philosophy of a limiting control rod pattern is a limiting control rod pattern exists when the reactor is operating at a MCPR limit equal to the unblocked Rod Withdrawal Error transient MCPR. The STS definition of a limiting control rod pattern is when a thermal limit is equal to 1.0. The current specifications allow the RBM to be bypassed for an indefinite period of time when the reactor is not operating on a limiting control rod pattern. The proposed specifications adopted from the STS, have more stringent requirements for the RBM operability. With one RBM inoperable and the reactor is not on a limiting control rod pattern, the RBM must be returned to the operable condition within 24 hours or the inoperable channel is to be placed in the trip condition. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

o) Proposed Specifications for the Economic Generation Control System, 3/4.3.N, are based on present provisions rewritten using an STS format. Since the present level of operability for the EGC System is retained by the proposed changes, the changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes do not 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.

The proposed changes for Dresden and Quad Cities Technical Specifications Section 3/4.3 are based on STS guidelines or later operating BWR plants' NRC accepted changes. These proposed changes have been reviewed for acceptability at the Dresden and Quad Cities Nuclear Power Stations considering similarity of system or component design versus the STS or later operating BWRs. No new modes of operation are introduced by the proposed changes, considering the acceptable operational modes in present specifications, the STS, or later operating BWRs. Surveillance requirements are changed to reflect improvements in technique, frequency of performance or operating experience at later plants. Proposed changes to action statements in many places add requirements that are not in the present technical specifications or adopt requirements that have been used successfully at other operating BWRs with designs similar to Dresden and Quad Cities. The proposed changes maintain at least the present level of operability. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

The proposed changes do not 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 proposed changes to Technical Specification Section 3/4.3 implement present requirements, or the intent of present requirements in accordance with the guidelines set forth in the STS. The proposed changes are intended to improve readability, usability, and the understanding of technical specification requirements while maintaining acceptable levels of safe operation. The proposed changes have been evaluated and found to be acceptable for use at Dresden and Quad Cities based on system design, safety analysis requirements and operational performance. Since the proposed changes are based on NRC accepted provisions at other operating plants that are applicable at Dresden and Quad Cities and maintain necessary levels of system, component or parameter operability, the proposed changes do not involve a significant reduction in 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 3/4.3

"REACTIVITY CONTROL"

ATTACHMENT 7

APPLICATION OF GENERIC LETTER 87-09
REVISION TO 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
3.3.A	1	1-2	18 hrs	NO	
	2	3-4	UNLIMITED	OK	
	3	5	UNLIMITED	OK	
3.3.B	-	1-2	UNLIMITED	OK	
3.3.C	1&2	1-2	UNLIMITED	OK	
	3&4	1-2	12 hrs	NO	
3.3.D	-	1-2	UNLIMITED	OK	
3.3.E	-	1-2	12 hrs	NO	
3.3.F	-	1-2	UNLIMITED	OK	
3.3.G	1	1-2	UNLIMITED	OK	
	2	5	UNLIMITED	OK	Shutdown is still Mode 5
3.3.H	1	1-2	UNLIMITED	OK	
	2	1-2	12 hrs	NO	
	3	5	UNLIMITED	OK	
3.3.I	1	1-2	UNLIMITED	OK	
	2	1-2	12 hrs	NO	
	3	5	UNLIMITED	OK	
3.3.J	-	1-3	36 hrs	NO	
3.3.K	1	1-2	36 hrs	NO	
	2	1-2	20 hrs	NO	
3.3.L	-	1-2	UNLIMITED	OK	
3.3.M	1,2&3	1	UNLIMITED	OK	
3.3.N	-	1	1 hr	NO	