Commonwealth Edison
1400 Opus Place
Downers Grove, Illinois 60515

June 20, 1991

Dr. Thomas E. Murley, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attn: Document Control Desk

Subject: Quad Cities Nuclear Power Station Units 1 and 2
Application for Amendment to Facility Operating
Licenses DPR-29 and DPR-30, Appendix A, Technical
Specifications
Technical Specification Upgrade Program
NRC Docket Nos. 50-254 and 50-265

Dr. Murley:

In 1991, Quad Cities Station initiated a formal program to enhance the Station's performance in various aspects of plant operation. The Performance Enhancement Program (PEP) was developed by assessing existing improvement plans, identifying potential weaknesses and prioritizing improvement actions. Necessary improvements to the Technical Specifications were identified as one of the Station top priority issue. Details of the goal and methodology for the Technical Specification Upgrade Program are provided in the Executive Summary section of the proposed amendment. The Program has been discussed with members of the NRR staff.

Pursuant to 10 CFR 50.90, Commonwealth Edison proposes to amend Appendix A, Technical Specifications to Facility Operating Licenses DPR-29 and DPR-30. The proposed amendment reflects Commonwealth Edison's efforts to upgrade existing Technical Specifications Sections 1.0, "Definitions" and 3.0/4.0 "Applicability". An overall description of the proposed amendment is also included in the Executive Summary.

The proposed amendment request for each Technical Specification section is provided as follows:

1. An Executive Summary of the Technical Specification Upgrade Program and the proposed amendment;

2. A summary of the changes;

3. A detailed description of the changes;

4. The proposed Technical Specification pages with the requested changes;

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5. The existing Technical Specification pages for DPR-29, are marked-up to reflect the appropriate changes to the existing Technical Specifications.

6. Commonwealth Edison's evaluation pursuant to 10 CFR 50.92(c) and 10 CFR 51.21; and,

7. The technical differences between the existing Unit 1 and Unit 2 Technical Specifications.

The proposed amendment has been reviewed and approved by Commonwealth Edison's on-site and off-site review in accordance with Company procedures.

The Technical Specification Upgrade Program proposes changes to each section of the existing Technical Specifications. As such, Commonwealth Edison requests that the proposed amendments be approved and issued as one document rather than as individual sections. All proposed changes to the Technical Specifications which are developed under this program will be submitted to the Commission by no later than September 30, 1991. Commonwealth Edison respectfully requests the NRC's approval of the upgraded Technical Specifications by July 1, 1992.

To the best of my knowledge and belief, the statements contained are true and correct. In some respects, these statements are not based on my personal knowledge but obtained information furnished by other Commonwealth Edison employees and consultants. Such information has been reviewed in accordance with Company practice and I believe it to be reliable.

Commonwealth Edison is notifying the State of Illinois of this application for amendment by transmitting a copy of this letter and its attachments to the designated State Official.

Please direct any questions concerning this proposed amendment to R. Stols at 708/515-7283.

Very truly yours,

Rita Stols

Rita Stols
Nuclear Licensing Administrator

Enclosures: A) Proposed Amendment to Technical Specification Section 1.1/2.2 "Reactor Coolant System"
B) Proposed Amendment to Technical Specification 3.0/4.0 "Applicability"

cc: A.B. Davis, Regional Administrator
L.N. Olshan, Project Manager
T.E. Taylor, Senior Resident Inspector
Illinois Department of Nuclear Safety

Signed before me on this 20th day of June, 1991.

By: _____

" OFFICIAL SEAL "
SANDRA C. LARA
NOTARY PUBLIC, STATE OF ILLINOIS
MY COMMISSION EXPIRES 6/25/94

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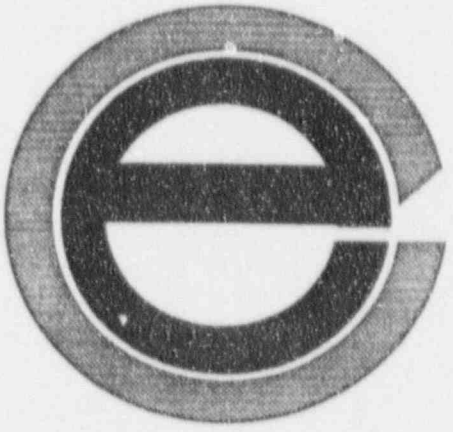
QUAD CITIES
NUCLEAR POWER STATION

TECHNICAL SPECIFICATION
UPGRADE PROGRAM



PROPOSED AMENDMENT
SECTION 1.0, "Definitions"

EXECUTIVE SUMMARY

Proposed Changes to TS 1.0

"DEFINITIONS"

EXECUTIVE SUMMARY

QUAD CITIES TECHNICAL SPECIFICATION UPGRADE PROGRAM

The Quad Cities Technical Specification Upgrade Program was conceptualized in response to lessons learned from the Dresden Diagnostic Evaluation Team inspection and the frequent need for Technical Specification interpretations. A comparison of the existing Quad Cities Technical Specification and, Standard Technical Specifications and later operating plants' Technical Specification provisions was conducted to identify potential improvements in clarifying requirements and to identify requirements which are no longer consistent with current industry practices. The comparison review identified approximately one-hundred and fifty suggested improvements. The Technical Specification Upgrade Program was not intended to be a complete adoption of the Standard Technical Specifications. Overall, the Quad Cities custom Technical Specifications provide for safe operation of the plant and, therefore, only a upgrade was deemed appropriate.

The comparison study revealed a mix of recommended upgrades which included the relaxation of certain existing Technical Specification requirements, the addition of surveillances, the removal of allowances which would no longer be allowed under new plant licensing, and better definition of appropriate action requirements in the event a Limiting Condition for Operation cannot be met. The Technical Specification Upgrade Program also implements NRC recommended line item improvements to the Technical Specifications which were issued under Generic Letters.

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

The Technical Specification Upgrade Program was identified as a Station top priority during the development of Quad Cities Station's Performance Enhancement Program (PEP). The Technical Specification Upgrade Program's goal is to provide a better tool to Station personnel to implement their responsibilities and to ensure Quad Cities Station is operated in accordance with current industry practices. The upgraded specifications provide for more safe and reliable operation of the plant. The program improves the operator's ability to use the Technical Specifications by more clearly defining Limiting Conditions for Operations 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 1.0, "Definitions"

All definitions have been arranged in alphabetical order and each definition has been assigned a specific numerical designation. Some changes are proposed to existing definitions and one existing definition, "Surveillance Interval" has been moved to Technical Specification Section 4.0.

New definitions have been added which apply to terms in current use in the Quad Cities Technical Specifications. New Table 1-1, "Surveillance Frequency Notation" and Table 1-2, "Operational Modes" follow the Standard Technical Specification (STS) format with notations and operational modes based on present Quad Cities allowances which are consistent with later operating plants and STS guidelines.

SUMMARY OF CHANGES

PROPOSED TS 1.0

'DEFINITIONS'

SUMMARY OF CHANGES

PROPOSED SPECIFICATION 1.0

DEFINITIONS

This part of the proposed amendment request contains changes to Quad Cities Units 1 and 2 Technical Specifications Section 1.0 on Definitions. Some changes are proposed to existing definitions, one existing definition ("Surveillance Interval") is moved to Technical Specification 4.0, and new definitions are added in accordance with STS guidelines. All the definitions are arranged in alphabetical order and each definition is assigned a specific numerical designator. New Tables are added for Operational Mode definitions and Surveillance Frequency definitions. The specific changes are described below.

Pages 1.0-1 through 1.0-6, DPR-29

Item 1:

The present definitions that are changed by this amendment request are as follows:

a. Present Definition 1.0.A, "Alteration of the Reactor Core", is changed to the STS and some later operating plants definition for "Core Alteration".

b. Present Definitions 1.0.E, 1.0.F, 1.0.G and 1.0.LL, "Instrument Calibration, Instrument Check, Instrument Functional Test and Channel Functional Test," respectfully, are replaced with STS versions of the definitions for Channel Calibration, Channel Check, and Channel Functional Test.

c. Present Definition 1.0.L, "Modes of Operation", is changed to "Mode Switch Interlocks". The new definition describes the reactor mode switch positions and associated interlocks.

d. Present Definition 1.0.M, "Operable", is changed to "Operable - Operability".

e. Present Definition 1.0.P, "Primary Containment Integrity", is modified to require all automatic containment isolation valves to be in compliance with the provisions of Specification 3.7.F.

f. Present Definition 1.0.Q, "Protective Instrumentation Definitions" is changed by adding the introductory phrase "Protective instrumentation definitions are as follows:".

g. Present Definition 1.0.X, "Secondary Containment Integrity" is modified to require the standby gas treatment system to be operable pursuant to Specification 3.7.K.

h. Present Definition 1.0.BB, "Critical Power Ratio (CPR)" is modified to STS wording with the incorporation of a generic reference for the critical power correlation in place of a reload specific correlation.

i. Present Definition 1.0.CC, "Minimum Critical Power Ratio," is clarified per STS guidelines to be the smallest CPR which exists in the core.

j. Present Definition 1.0.GG, "Fraction of Rated Power (FRP)," is clarified by stating that the core thermal power is the measured core power.

k. Present Definitions 1.0.JJ, "Process Control Program," and 1.0.KK, "Offsite Dose Calculation Manual (ODCM)," are modified per Generic Letter 89/01.

l. An administrative change is proposed to many of the present definitions in order to promote consistency in presentation of the definitions. The definition term is repeated at the beginning of each definition paragraph.

Item 2:

New definitions are added for the following terms:

a. Action

b. Average Planar Exposure

c. Average Planar Linear Heat Generation Rate

d. Identified Leakage

e. Limiting Control Rod Pattern

f. Linear Heat Generation Rate (LHGR)

g. Operational Mode

h. Pressure Boundary Leakage

i. Physics Tests

j. Shutdown Margin

k. Site Boundary

l. Unidentified Leakage

m. Unrestricted Area

Item 3:

New Table 1-1, "Surveillance Frequency Notation", is added to the technical specifications.

Item 4:

New Table 1-2, "Operational Modes" is added to the technical specifications.

Item 5:

The provisions of present Definition 1.0.DD, "Surveillance Interval", are modified and moved to proposed specification 4.0.B. The present provision that requires the total maximum combined surveillance interval time for any 3 consecutive surveillance intervals not to exceed 3.25 times the specified interval, is deleted.

Item 6:

With the inclusion of proposed Table 1-2 on Operational Modes, the present Definition 1.0.Y, "Shutdown", is no longer needed and is deleted.

DESCRIPTION OF CHANGES

PROPOSED TS 1.0

'DEFINITIONS'

DESCRIPTION OF PROPOSED AMENDMENT REQUEST

PROPOSED SPECIFICATION 1.0

DEFINITIONS

The changes proposed in this amendment request for Section 1.0 of the Quad Cities Technical Specifications are made to provide a more complete listing of the definition terms, to provide improvements to certain STS definitions based on later operating BWR plants' terms, and to allow implementation of Generic Letter 89/01.

An item by item description of the proposed changes requested is provided below. The Summary of Changes section can be referred to in order to reference back to a given change and its affected pages.

Item 1

Item 1 addresses the proposed changes to present definitions. The present Definition 1.0.A for "Alteration of the Reactor Core" is changed to the STS and later operating definition of "Core Alteration." The proposed change to present Definition 1.0.A clarifies the intent of present provisions by stating that the definition applies only when the reactor vessel head is removed and fuel is in the vessel. The proposed definition of Core Alteration provides an exclusion for the undervessel replacement of incore instrumentation. This exclusion from the provisions of the Core Alteration definition has been adopted at some later operating plants and provides necessary relief from the restrictions placed on plant operations during actual Core Alterations.

The present definitions for Instrument Calibration, Instrument Check, Instrument Functional Test and Channel Functional Test (Radiation Monitor) are proposed to be replaced with STS definitions for Channel Calibration, Channel Check, and Channel Functional Test. This change will delete the present outdated terminology that limits the definitions to the "Instrument" and does not include the entire "Channel."

Present Definition 1.0.L, "Modes of Operation", is proposed to be changed to "Mode Switch Interlocks". A new definition for Operational Modes is added which references new Table 1-2 to define the reactor mode switch positions and reactor coolant temperatures associated with Operational Modes.

The present definition for "Operable" is expanded to include "Operability" as part of the Operable terminology. This change follows STS guidelines and provides for consistency in word usage and definitions throughout the technical specifications.

Present Definition 1.0.P, "Primary Containment Integrity" is proposed to be changed to require all automatic containment isolation valves to be in compliance with the provisions of Specification 3.7.F instead of the present requirement for all valves to be operable or deactivated in the isolated condition. The proposed change provides consistency between the definition of Primary Containment Integrity and proposed Specification 3.7.F operability requirements for primary containment isolation valves. Compliance with the provisions of Specification 3.7.F actions will satisfy the isolation valve operability requirements for the definition of Primary Containment Integrity.

The proposed change to present Definition 1.0.Q, "Protective Instrumentation Definitions" is administrative in nature and adds an introductory phrase to improve readability.

Present Definition 1.0.X, "Secondary Containment Integrity", is proposed to be changed to require the Standby Gas Treatment System to be operable pursuant to Specification 3.7.K requirements instead of the present provision that the Standby Gas Treatment System be operable. The proposed change will allow the Standby Gas Treatment System to be in compliance with the Action Statements of Specification 3.7.K and meet the intent of operability for Secondary Containment Integrity. Present provisions of Definition 1.0.X could be interpreted to require that both trains of the Standby Gas Treatment System be operable to meet Secondary Containment Integrity requirements. The proposed change follows STS and later operating plant practices of allowing system operability to be determined by specific system specifications.

The proposed change to present Definition 1.0.BB, "Critical Power Ratio (CPR)," is made to preclude future Technical Specification Definition revisions due to minor changes in the fuel manufacturer's critical power correlations. The reload specific "GEXL" is replaced by "applicable NRC-approved critical power correlation". The proposed change follows STS and the Hope Creek Generating Station amendment published in 4870 FR, Vol. 56, No. 25, Wednesday, February 6, 1991.

The proposed change to present Definition 1.0.CC, "Minimum Critical Power Ratio," is made to clarify present wording. The present definition refers to the most limiting fuel assembly in the core; whereas, the proposed change rewords this statement to refer to the MCPR as the smallest CPR which exists in the core. This clarification does not change the technical meaning of this definition.

The change proposed to present Definition 1.0.GG, "Fraction of Rated Power," adds the clarification that the core thermal power is the measured core thermal power and not the actual core power. This change does not change present intent but does clarify the wording for users of the Quad Cities Technical Specification. Generic Letter 89/01 was issued on January 31, 1989, and contains

the guidance for relocating the Technical Specification provisions for Radiological Effluents to the Offsite Dose Calculation Manual and the Process Control Program. The proposed changes to present Definitions 1.0.JJ, "Process Control Program," and 1.0.KK, "Offsite Dose Calculation Manual," are made in order to implement the guidance of GL 89/01. Changes to other sections of the Quad Cities Technical Specifications in order to implement GL 89/01 are reflected in the changes proposed for those sections.

The proposed administrative change to the definitions will provide consistency of presentation by starting each definition paragraph with the defined term.

Item 2

Twelve of the newly proposed definitions are taken from STS guidelines and are added to the Quad Cities Technical Specifications to provide a more complete list of defined terms. These additions represent new restrictions or clarifications not presently in the Quad Cities Technical Specifications.

Item 3

Item 3 addresses the addition of Table 1-1, "Surveillance Frequency Notation", to the Quad Cities Technical Specifications. STS guidelines, current plant technical specification allowances and current plant operating practices were used to develop proposed Table 1-1.

The STS provides a definition of Refueling Cycle as at least once per 18 months (550 days). Current plant interpretation of Refueling Cycle is provided by present Definition 1.0.O, "Operating Cycle". The present definition of "Operating Cycle" is the interval between the end of one refueling outage for a particular unit and the end of the next subsequent refueling outage for the same unit. The present definition of "Operating Cycle" is retained in proposed Table 1-1 to define the surveillance frequency for a refueling cycle. In order to provide a surveillance frequency notation for those surveillances that are to be performed at 18 month intervals, the letter "E" is added to indicate a surveillance frequency of at least once per 18 months (550 days).

Item 4

Proposed Table 1-2, "Operational Modes", implements present Quad Cities Technical Specification allowances, STS provisions, and later operating plant specification allowances. Present Definition 1.0.L uses the terminology "Modes of Operation" instead of the STS terminology of "Operational Conditions". Since the present users of the technical specifications prefer the "Mode" terminology, Quad Cities has elected to use the terminology "Operational Modes" in Table 1-2 and throughout the technical specifications. Present Quad Cities temperature conventions are

retained for Operational Modes 1, 2, 3, and 4. STS guideline of $\leq 140^{\circ}F$ is used for Operational Mode 5. Proposed Table 1-2 Notes (a) and (c) are taken from STS guidelines and provide a definition of the Operational Mode of "Refueling" and allows movement of the reactor mode switch to the Run or Startup/Hot Standby positions to test switch interlocks.

Proposed Table 1-2 Note (b) is taken from later operating plants specifications and addresses the condition where there is no fuel in the reactor vessel. In this condition, the reactor is considered not to be in any Operational Mode and the reactor mode switch may be in any position or inoperable. This table note is

necessary in order to provide operational flexibility in the case where all fuel is removed from the vessel and mode switch operability could restrict planned maintenance or other activities.

Proposed Notes (d) and (e) for Table 1-2 are taken from STS guidelines. Proposed Note (d) allows the reactor mode switch to be placed in the Refuel position while the plant is in Cold Shutdown to allow a single control rod to be removed from the reactor pressure vessel in accordance with proposed Specification 3.10.D. Proposed Note (e) allows the reactor mode switch to be placed in the Refuel position while the plant is in Hot Shutdown or Cold Shutdown while a single control rod is being recoupled or withdrawn provided that the one-rod-out interlock in OPERABLE.

Item 5

Present Definition 1.0.DD, "Surveillance Interval," contains limitations on maximum allowable extensions to surveillance intervals. These provisions are contained in Specification 4.0.B of the STS and this proposed change moves the applicable provisions of present Definition 1.0.DD to proposed Specification 4.0.B considering the implementation of Generic Letter 89-14. On August 21, 1989, the NRC Staff issued Generic Letter 89-14 which removes the 3.25 limit from Specification 4.0.2 (4.0.B in Quad Cities Technical Specifications). The NRC Staff 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. As discussed in the proposed changes to Section 3.0/4.0 of the Quad Cities Technical Specifications, this provision of the Generic Letter will be adopted with the resulting deletion of present Definition 1.0.DD.

Item 6

Present Definition 1.0.Y, "Shutdown," contains the Operational Modes of Hot Shutdown and Cold Shutdown. Since the Operational Modes are proposed to be defined in new Table 1-2, this definition is no longer needed and is deleted.

PROPOSED TECH SPEC

TS 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

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

AVERAGE PLANAR EXPOSURE

1.2 The AVERAGE PLANAR EXPOSURE 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

1.3 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 RATES 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 CALIBRATION

1.4 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

1.5 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 CHANNELS measuring the same parameter.

CHANNEL FUNCTIONAL TEST

1.6 A CHANNEL FUNCTIONAL TEST shall be:

a. Analog CHANNELS - 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 CHANNELS - 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

1.7 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 undervessel replacement) of the SRMs, IRMs, TIPs, LPRMs, or special movable detectors is not considered a CORE ALTERATION. Suspension of CORE ALTERATIONS shall not preclude completion of the movement of a component to a safe conservative position.

CORE OPERATING LIMITS REPORT

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

CRITICAL POWER RATIO (CPR)

1.9 The CRITICAL POWER RATIO is the ratio of that assembly power which causes some point in the assembly to experience TRANSITION BOILING to the assembly power at the reactor condition of interest as calculated by application of the GEXL correlation (Reference NEDO-10958).

DOSE EQUIVALENT I-131

1.10 DOSE EQUIVALENT I-131 is 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."

DUAL LOOP OPERATION (DLO)

1.11 DUAL LOOP OPERATION is REACTOR POWER OPERATION with both recirculation pumps running.

FRACTION OF LIMITING POWER DENSITY (FLPD)

1.12 The FRACTION OF LIMITING POWER DENSITY is the ratio of the LINEAR HEAT GENERATION RATE (LHGR) existing at a given location to the design LHGR for that bundle type.

FRACTION OF RATED POWER (FRP)

1.13 The FRACTION OF RATED POWER is the ratio of measured core thermal power to RATED THERMAL POWER of 2511 MWth.

HOT STANDBY

1.14 HOT STANDBY means operation with the reactor critical, system pressure less than 1060 psig, the main steam isolation valves closed, and thermal power not exceeding 15%.

IDENTIFIED LEAKAGE

1.15 IDENTIFIED LEAKAGE shall be:

a. Leakage into 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 drywell 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.

IMMEDIATE

1.16 IMMEDIATE means that the required action will be initiated as soon as practicable considering the safe operation of the unit and the importance of the required action.

LIMITING CONDITIONS FOR OPERATION (LCO)
1.17 The LIMITING CONDITIONS FOR OPERATION are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When these conditions are not met, the plant shall be shut down or any remedial action permitted by the technical specifications shall be followed until the condition can be met.

LIMITING CONTROL ROD PATTERN
1.18 A LIMITING CONTROL ROD PATTERN 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.

LIMITING SAFETY SYSTEM SETTING (LSSS)
1.19 The LIMITING SAFETY SYSTEM SETTINGs are the settings of instrumentation which initiates the automatic protective action at a level such that the SAFETY LIMITS will not be exceeded. The region between the SAFETY LIMIT and these settings represents margin, with normal operation lying below these settings. The margin has been established so that with proper operation of the instrumentation, the SAFETY LIMITS will never be exceeded.

LINEAR HEAT GENERATION RATE (LHGR)
1.20 LINEAR HEAT GENERATION RATE 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
1.21 A LOGIC SYSTEM FUNCTIONAL TEST means a test of all relays and contacts of a logic circuit from sensor to activated device to ensure all components are OPERABLE per design intent. Where possible, action will go to completion, i.e., pumps will be started and valves opened. 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)
1.22 The MAXIMUM FRACTION OF LIMITING POWER DENSITY is the highest value existing in the core of the FRACTION OF LIMITING POWER DENSITY (FLPD).

MEMBERS OF THE PUBLIC

1.23 MEMBERS OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the utility, its contractors, or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational, or other purposes not associated with the plant.

MINIMUM CRITICAL POWER RATIO (MCPR)

1.24 The MINIMUM CRITICAL POWER RATIO shall be the smallest CPR which exists in the core.

Mode Switch Interlocks

1.25 The reactor mode switch selects the proper interlocking for the operating or shutdown condition of the plant. Following are the reactor mode switch positions and interlocks provided:

a. SHUTDOWN - In this position, a reactor scram is initiated, power to the control rod drives is removed, and the reactor protection trip systems have been deenergized for 10 seconds prior to permissive for manual reset.

b. REFUEL - In this position, interlocks are established so that one control rod only may be withdrawn when flux amplifiers are set at the proper sensitivity level and the refueling crane is not over the reactor. Also the trips from the turbine control valves, turbine stop valves, main steam isolation valves, and condenser vacuum are bypassed. If the refueling crane is over the reactor, all rods must be fully inserted and none can be withdrawn.

c. STARTUP/HOT STANDBY - In this position, the reactor protection scram trips, initiated by condenser low vacuum and main steamline isolation valve closure, are bypassed, the low pressure main steamline isolation valve closure trip is bypassed, and the reactor protection system is energized, with IRM and APRM neutron monitoring system trips and control rod withdrawal interlocks in service.

d. RUN - In this position the reactor system pressure is at or above 825 psig, and the reactor protection system is energized, with APRM protection and RBM interlocks in service (excluding the 15% high flux scram).

OFFSITE DOSE CALCULATION MANUAL (ODCM)

1.26 The OFFSITE DOSE CALCULATION MANUAL 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 6.2 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

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

OPERATING

1.28 OPERATING means that a system, subsystem, train, component or device is performing its intended functions in its required manner.

OPERATING CYCLE

1.29 OPERATING CYCLE is the interval between the end of one REFUELING OUTAGE for a particular unit and the end of the next subsequent REFUELING OUTAGE for the same unit.

OPERATIONAL MODE

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

PHYSICS TESTS

1.31 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 3 of the FSAR, 2) authorized under the provisions of 10 CFR 50.59, or 3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

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

PRIMARY CONTAINMENT INTEGRITY

1.33 PRIMARY CONTAINMENT INTEGRITY means that the drywell and pressure suppression chamber are intact and all of the following conditions are satisfied:

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

b. At least one door in each air lock is closed and sealed.

c. All automatic containment isolation valves are in compliance with the provisions of Specification 3.7.F.

d. All blind flanges and manways are closed.

PROCESS CONTROL PROGRAM (PCP)

1.34 The PROCESS CONTROL PROGRAM 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.

Protective Instrumentation Definitions

1.35 Protective instrumentation definitions are as follows:

a. CHANNEL - A CHANNEL is an arrangement of a sensor and associated components used to evaluate plant variables and produce discrete outputs used in logic. A CHANNEL terminates and loses its identity where individual CHANNEL outputs are combined in a logic.

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

c. PROTECTIVE ACTION - An action initiated by the protection system when a limit is reached. A PROTECTIVE ACTION can be at the CHANNEL or system level.

d. PROTECTIVE FUNCTION - A system protective action which results from the PROTECTIVE ACTION of the CHANNELS monitoring a particular plant condition.

RATED NEUTRON FLUX

1.36 RATED NEUTRON FLUX is the neutron flux that corresponds to a steady-state power level of 2511 thermal megawatts.

RATED THERMAL POWER

1.37 RATED THERMAL POWER means a steady-state power level of 2511 thermal megawatts.

REACTOR POWER OPERATION

1.38 REACTOR POWER OPERATION is any operation with the mode switch in the STARTUP/HOT STANDBY or RUN position with the reactor critical and above 1% RATED THERMAL POWER.

REACTOR VESSEL PRESSURE

1.39 REACTOR VESSEL PRESSURES listed in the Technical Specifications, unless otherwise indicated, are those measured by the reactor vessel steam space detector.

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

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

SAFETY LIMIT
1.42 The SAFETY LIMITS are limits below which the reasonable maintenance of the cladding and primary system are assured. Exceeding such a limit is cause for unit shutdown, and review by the NRC before resumption of unit operation. Operation beyond such a limit may not in itself result in serious consequences, but it indicates an operational deficiency subject to regulatory review.

SECONDARY CONTAINMENT INTEGRITY
1.43 SECONDARY CONTAINMENT INTEGRITY means that the reactor building is intact and the following conditions are met:

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

b. The standby gas treatment system is OPERABLE pursuant to Specification 3.7.K.

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

SHUTDOWN MARGIN
1.44 SHUTDOWN MARGIN 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. With a control rod not capable of being fully inserted, the reactivity worth of this control rod must be accounted for in the determination of SHUTDOWN MARGIN.

SIMULATED AUTOMATIC ACTUATION
1.45 SIMULATED AUTOMATIC ACTUATION means applying a simulated signal to the sensor to actuate the circuit in question.

SINGLE LOOP OPERATION (SLO)
1.46 SINGLE LOOP OPERATION is REACTOR POWER OPERATION with one recirculation pump running.

SITE BOUNDARY
1.47 The SITE BOUNDARY shall be that line beyond which the land is neither owned, nor otherwise controlled by the licensee.

SOURCE CHECK
1.48 SOURCE CHECK is the qualitative assessment of instrument response when the sensor is exposed to a radioactive source.

TRANSITION BOILING
1.49 TRANSITION BOILING means the regime between nucleate and film boiling. TRANSITION BOILING is the regime in which both nucleate and film boiling occur intermittently, with neither type being completely stable.

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

UNRESTRICTED AREA
1.51 An UNRESTRICTED AREA shall be any area at or beyond the SITE BOUNDARY access to which is not controlled by the licensee for purposes of protection of MEMBERS OF THE PUBLIC from exposure to radiation and radioactive materials, or any area within the SITE BOUNDARY used for residential quarters or for industrial, commercial, institutional, and/or recreational purposes.

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.	Refueling cycle	R	At least once per OPERATING CYCLE
9.	18 Months	E	At least once per 18 months (550 days)
10.	Startup	S/U	Prior to each reactor startup
11.	Not Applicable	N.A.	Not applicable

TABLE 1-2

OPERATIONAL MODES

MODES	MODE SWITCH POSITION (b)	AVERAGE REACTOR COOLANT TEMPERATURE
1. RUN	RUN	Any temperature
2. STARTUP	STARTUP/HOT STANDBY	Any temperature
3. HOT SHUTDOWN	SHUTDOWN (c)(e)	> 212° F
4. COLD SHUTDOWN	SHUTDOWN (c)(d)(e)	< 212° F
5. REFUELING (a)	SHUTDOWN or REFUEL (c)	< 140° F

TABLE NOTATIONS

a. Fuel in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

b. 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.

c. The reactor mode switch may be placed in the RUN, STARTUP/HOT STANDBY, or REFUEL position to test the switch interlock functions and related instrumentation provided that the control rods are verified to remain fully inserted by a second licensed operator or other technically qualified engineer of the unit technical staff.

d. 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.

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

EXISTING TECH SPEC

TS 1.0

'DEFINITIONS'