Proposed Change RTS-156A
to the
Duane Arnold Energy Center
Technical Specifications

The holders of license DPR-49 for the Duane Arnold Energy Center propose to amend Appendix A (Technical Specifications) to said license by deleting current pages and replacing them with attached, new pages. A List of the Affected Pages is given below.

The following list of proposed changes is only administrative. The changes being made can be divided into three categories: Changes made to clarify existing wording, Changes made to update old references, and Changes made to correct typos.

Changes being made to clarify existing wording:

(1) Section 3.2.C.2 is changed to clarify the time limit during which the minimum number of Rod Block Monitor channels can be reduced. The change does not constitute a change in the substance of the Specification.

(2) A title ("Instrumentation that Initiates Control Rod Blocks") has been added to Table 3.2-C to clarify what it contains.

(3) Section 3.7.A.5.b is changed to add "with the intent of commencing power operation" so in the event the reactor is put in the run mode to complete startup testing, a LCO does not have to be entered.

(4) Sections 3.7.A.6.a and 3.7.A.6.b are changed to clarify the action statements. The current specifications simply state that "the reactor must be taken out of power operation". The proposed change will require a hot shutdown within 24 hours if the operability requirements cannot be met within the grace period.

(5) Section 3.8 bases is changed to include requirements for operation with inoperable 24V batteries. A new paragraph is added to address what happens with these batteries inoperable.

(6) Section 1.1 is changed to clarify "less than" and "less than or equal to" for core thermal power limit and core flow. Signs were inconsistent and this has been changed.

(7) Tables 3.2-A and 3.2-B have been changed to allow several of the setpoints to be more easily set with either (1) an increase in protection, or (2) no change in protection. Current setpoints are listed as one number, and the proposed change will add "less than or equal to", or "greater than or equal to" to each setting.

(8) Section 3.12.C is changed to delete the phrase "during reactor power operation" in reference to the action statement for MCPR limits being executed. This is being done to avoid interpretation problems.

(9) Table 3.2-B is changed to add delta (Δ) to the HPCI Room differential temperature set point. Explanation of four instrument channels for HPCI and RCIC room high temperature trips are also being added.

(10) Page vi, reference to Table 3.6-1 is deleted. Table 3.6-1 was deleted by Amendment 56.

(11) Page vii, Figure 4.8.C-1 reference is added to List of Figures and reference to "deleted" figures is dropped.

(12) Section 4.6.B.1.h is changed to replace incorrect Title.

(13) Page 3.1-21, Section 3.1 bases is revised to add "<" before the ARM reading of "5" for insertion of the IRMs. The "less than" sign was originally included in the bases, but was inadvertently deleted by Amendment 14.

(14) Page 3.3-6, Section 4.3.C.1, is revised to add "to 50% rod density" in order to provide the proper range to indicate which rods are to be scram time tested. The Technical Specifications originally included this statement, but it was inadvertently deleted by Amendment 54.

(15) Table 3.2-B, HPCI Turbine Steam Line High Flow trip settings are being changed. Amendment 26 originally contained this change, but Amendment 28 inadvertently deleted it.

(16) Bases Section 3.5.E is changed to explain maximum allowable repair time for RCIC.

Changes being made to update old references:

(1) Section 3.6.B.2.a is changed to delete reference to the Limits on Conductivity during the first occasion on which reactor coolant water temperature, as a result of nuclear heatup, reached 375°F. This occasion has already passed so that portion of the Specification is no longer applicable.

(2) Sections 3.8.B.4 and 6.11 are changed to replace references to the "AEC" with "NRC".

(3) Sections 6.8.1 and 6.8.4 are changed to replace "Preparedness Plan" with "Emergency Plan". "Quarterly" is taken out of Section 6.8.4 because the Emergency Plan does not require quarterly drills.

(4) Section 6.11.1.b is changed to replace "prior to March 1" with "within 60 days of January 1" to make consistent with the rest of the reporting procedures.

(5) Page vi of the Table of Contents is changed to remove the reference to Table 6.9-1. This table no longer exists.

(6) Section 4.3.C.2 and 4.3.C.3 are now deleted. These sections were only valid prior to Cycle 6.

The Bases for Section 3.3 and 4.3 is revised to delete the reference to the tests described in Sections 4.3.C.2 and 4.3.C.3.

(7) Section 3.3.B.5 has been changed to delete the phrase "designated qualified personnel". The responsibilities of the personnel are adequately defined elsewhere (p. 3.3-17).

(8) Tables 3.2-H and 4.2-H are added to the Table of Contents and List of Tables.

(9) Section 3.3.2.e is changed to replace reference to DAEC Chief Engineer with Plant Superintendent - Nuclear.

(10) Page 1.0-1, definition 1. is changed to include the reference to Nuclear Regulatory Commission rather than Atomic Energy Commission.

(11) Table 3.2-B is changed to reference Reactor Low Level to current Top of Active Fuel.

(12) The following pages are changed to update references to the FSAR.

1.0-7	3.3-12	3.7-11
1.1-14	3.3-13	3.7-12
1.2-4	3.4-7	3.7-32a
3.1-15	3.5-14	3.7-39
3.1-19	3.5-20	3.7-44
3.2-36	3.6-2	3.7-49
3.2-37	3.6-16	5.1-1
3.2-38	3.6-27	5.3-1
3.2-39	3.6-33	5.4-1
3.3-8	3.6-37	

Changes being made to correct typographical errors:

(1) Surveillance requirement 4.1.A.3 is changed to replace reference 2.1.B to 2.1.A.3. Currently it references the wrong table.

(2) Table 3.1-1 is changed to read "$\leq$ 120/125 of Full Scale" instead of "$\leq$ 120/125 of Fuel Scale".

(3) Section 6.8.2 is changed to replace 6.3.1 with 6.8.1.

(4) Sections 6.7.3 and 6.7.2 are changed to replace reference to 6.12 with 6.11, the correct reference.

(5) Bases Section 3.2 is changed to replace "6 times normal background" to "3 times normal background", the correct value.

LIST OF AFFECTED PAGES

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TECHNICAL SPECIFICATIONS

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TECHNICAL SPECIFICATIONS

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1.0 DEFINITIONS

The succeeding frequently used terms are explicitly defined so that a uniform interpretation of the specifications may be achieved.

1. SAFETY LIMIT

The safety limits are limits below which the reasonable maintenance of the cladding and primary systems are assured. Exceeding such a limit requires unit shutdown and review by the Nuclear Regulatory Commission 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.

2. LIMITING SAFETY SYSTEM SETTING (LSSS)

The limiting safety system settings are settings on instrumentation which initiate the automatic protective action at a level such that the safety limits will not be exceeded. These settings take into consideration the instrumentation tolerances and the instruments are required to be periodically calibrated as specified in these Technical Specifications. The limiting safety system setting plus the tolerance of the instrument as given in the system design control document gives the limiting trip point for operation. This additional margin has been established so that with proper operation of the instrumentation the safety limits will never be exceeded. The inequality sign which may be given merely signifies the preferred direction of operational trip setting.

3. LIMITING CONDITIONS FOR OPERATION (LCO)

The limiting conditions specify the minimum acceptable levels of system performance necessary to assure safe startup and operation of the facility. When these conditions are met, the plant can be operated safely and abnormal situations can be safely controlled.

When a system, subsystem, train, component or device is determined to be inoperable solely because its emergency power source is inoperable, or solely because its normal power source is inoperable, it may be considered OPERABLE for the purpose of satisfying the requirements of its applicable Limiting Condition for Operation, provided: (1) its corresponding normal or emergency power source is OPERABLE; and (2) all of its redundant system(s), subsystem(s), train(s), component(s) and devices(s) are OPERABLE, or likewise satisfy the requirements of this specification.

4. DELETED

22. Instrumentation - Continued

h. Protective Function - A system protective action which results from the protective action of the channels monitoring a particular plant condition.

i. Simulated Automatic Actuation - Simulated automatic actuation means applying a simulated signal to the sensor to actuate the circuit in question.

j. Logic - A logic is an arrangement of relays, contacts, and other components that produces a decision output.

   1) Initiating - A logic that receives signals from channels and produces decision outputs to the actuation logic.

   2) Actuation - A logic that receives signals (either from initiating logic or channels) and produces decision outputs to accomplish a protective action.

k. Primary Source Signal - The first signal, which by plant design, should initiate a reactor scram for the subject abnormal occurrence (see Updated FSAR Chapters 7 and 15).

23. FUNCTIONAL TEST

A functional test is the manual operation of initiation of a system, subsystem, or component to verify that it functions within design tolerances (e.g., the manual start of a core spray pump to verify that it runs and that it pumps the required volume of water).

24. SHUTDOWN

The reactor is in a shutdown condition when the reactor mode switch is in the shutdown mode position and no core alterations are being performed.

25. ENGINEERED SAFEGUARD

An engineered safeguard is a safety system the actions of which are essential to a safety action required in response to accidents.

SAFETY LIMIT

1.1 FUEL CLADDING INTEGRITY

Applicability:

Applies to the inter-related variables associated with fuel thermal behavior.

Objective:

To establish limits which ensure the integrity of the fuel cladding.

Specifications:

A. Reactor Pressure > 785 psig and Core Flow > 10% of Rated

The existence of a minimum critical power ratio (MCPR) less than 1.07 shall constitute violation of the fuel cladding integrity safety limit.

B. Core Thermal Power Limit (Reactor Pressure $\leq$ 785 psig or Core Flow $\leq$ 10% of Rated

When the reactor pressure is $\leq$785 psig or core flow is less than or equal to 10% of rated, the core thermal power shall not exceed 25 percent of rated thermal power.

LIMITING SAFETY SYSTEM SETTING

2.1 FUEL CLADDING INTEGRITY

Applicability:

Applies to trip settings of the instruments and devices which are provided to prevent the reactor system safety limits from being exceeded.

Objective:

To define the level of the process variables at which automatic protective action is initiated to prevent the fuel cladding integrity safety limits from being exceeded.

Specifications:

The limiting safety system settings shall be as specified below:

A. Neutron Flux Trips

1. APRM High Flux Scram When In Run Mode.

For operation with the fraction of rated power (FRP) greater than or equal to the maximum fraction of limiting power density (MFLPD), the APRM scram trip setpoint shall be as shown on Figure 2.1-1 and shall be:

$S \leq (0.66W + 54)$

with a maximum setpoint of 120% rated power at 100% rated recirculation flow or greater.

1.1 BASES: FUEL CLADDING INTEGRITY

A. Fuel Cladding Integrity Limit at Reactor Pressure > 785 psig and Core Flow > 10% of Rated

The fuel cladding integrity safety limit is set such that no fuel damage is calculated to occur if the limit is not violated. Since the parameters which result in fuel damage are not directly observable during reactor operation the thermal and hydraulic conditions resulting in a departure from nucleate boiling have been used to mark the beginning of the region where fuel damage could occur. Although it is recognized that a departure from nucleate boiling would not necessarily result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. However, the uncertainties in monitoring the core operating state and in the procedure used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore the fuel cladding integrity safety limit is defined as the critical power ratio in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition considering the power distribution within the core and all uncertainties.

The Safety limit MCPR is generically determined in Reference 1.

TABLE 3.[illegible] (Continued)

INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

Minimum No. of Operable Instrument Channels Per Trip System (1)	Trip Function	Trip Level Setting	Number of Instrument Channels Provided by Design	Remarks
2	Suppression Chamber HPCI Suction Level	$\leq$ 5" above normal water level	2 Instrument Channels	Transfers HPCI pump suction to suppression chamber
1	RCIC Turbine High Flow	+ 110" $\pm$ 5" H_2O (2)	2 Instrument Channels	
2	RCIC Turbine Equipment Room High Ambient Temperature	$\leq$ 175°F (2)	4 Instrument Channels	2 Channels for auto isolation and 2 channels for annunciation
2	RCIC Vent High Differential Temperature	$\leq$ Δ 50°F (2)	4 Instrument Channels	2 Channels for auto isolation and 2 channels for annunciation
2	RCIC Steam Line Low Pressure	100 > P > 50 psig (2)	4 Instrument Channels	
1	HPCI Turbine Steam Line High Flow	+53" H_2O (outboard Instr.) +99" H_2O (Inboard Instr.) (3)	2 Instrument Channels	
2	Suppresion Pool Area High Ambient Temperature	$\leq$ 150°F	4 Instrument Channels	
2	Suppression Pool Area High Differential Temperature	$\leq$ Δ50°F	4 Instrument Channels	
1	HPCI Leak Detection Time Delay	$\leq$ 15 min.	2 Instrument Channels	

2.1 BASES: LIMITING SAFETY SYSTEM SETTINGS RELATED TO FUEL CLADDING INTEGRITY

The abnormal operational transients applicable to operation of the Duane Arnold Energy Center have been analyzed throughout the spectrum of planned operating conditions up to the thermal power condition of 1658 MWt. The analyses were originally based upon plant operation in accordance with the operating map given in Figure 3.7-1 of the initial FSAR. Subsequent analyses have been based on updated maps, such as Figure 4.4-1 of the Updated FSAR. The current operating map is included as Figure 1.1-1 of these Technical Specifications. In addition, 1658 MWt is the licensed maximum power level of the Duane Arnold Energy Center, and this represents the maximum steady state power which shall not knowingly be exceeded.

Conservatism is incorporated in the transient analysis in estimating the controlling factors, such as void reactivity coefficient, control rod scram worth, scram delay time, peaking factors, and axial power shapes. These factors are selected conservatively with respect to their effect on the applicable transient results as determined by the current analysis mode. Conservatisms incorporated into the transient analysis is documented in Reference 1.

The absolute value of the void reactivity coefficient used in the analysis is conservatively estimated to be about 25% greater than the nominal maximum absolute value expected to occur during the core lifetime. The scram worth used has been derated to be equivalent to approximately 80% of the total scram worth of the control rods. The scram delay time and rate of rod insertion allowed by the analyses are conservatively set equal to the longest delay and slowest insertion rate acceptable by Technical Specifications. The design pressure (120% x 1150 = 1380 psig; 120% x 1325 = 1590 psig).

The Updated FSAR (Section 15.2.2.2) states that the turbine trip from high power without bypass can be the most severe abnormal operational transient resulting directly in a reactor coolant system pressure increase. The reactor vessel pressure code limit of 1375 psig, given in Subsection 5.2.2 of the Updated FSAR, is well above the peak pressure produced by the worst overpressure transient above. Thus, the pressure safety limit is well above the peak pressure that can result from reasonably expected overpressure transients.

A safety limit is applied to the Residual Heat Removal System (RHRS) when it is operating in the shutdown cooling mode. At this time it is included in the reactor coolant system.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT

3.1 REACTOR PROTECTION SYSTEM

Applicability:

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

Objective:

To assure the operability of the reactor protection system.

Specification:

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

As a minimum, the reactor protection system instrumentation channels of Table 3.1-1 shall be operable with response times as shown in Table 3.1-2.

4.1 REACTOR PROTECTION SYSTEM

Applicability:

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

Objective:

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

Specification:

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

.2 Response time measurements (from actuation of sensor contacts or trip point to de-energization of scram solenoid relay) are not part of the normal instrument calibration. The reactor trip system response time of each reactor trip function shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least one logic train such that both logic trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific reactor trip function.

.3 Daily during reactor power operation, the MFLPD and the FRP shall be checked and the APRM SCRAM and APRM Rod Block settings given by equations in Specification 2.1.A.1 and 2.1.A.3 shall be calculated if the MFLPD exceeds the FRP.

.4 When it is determined that a channel has failed in the unsafe condition, the other RPS channels that monitor the same variable shall be functionally

3.1-1

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENT

Minimum No. of Operable Instrument Channels for Trip System (1)	Trip Function	Trip Level Setting	Modes in Which Function Must be Operable			Number of Instrument Channels Provided by Design	Action (1)
			Refuel (6)	Startup	Run		
1	Mode Switch in Shutdown		X	X	X	1 Mode Switch (4 sections)	A
1	Manual Scram		X	X	X	2 Instrument Channels	A
2	IRM High Flux	≤ 120/125 of Full Scale	X	X	(5)	6 Instrument Channels	A
2	IRM Inoperative		X	X	(5)	6 Instrument Channels	A
2	APRM High Flux	(.66W+54) (FRP/MFLPD) (11) (12)			X	6 Instrument Channels	A or B
2	APRM Inoperative	(10)	X	X	X	6 Instrument Channels	A or B
2	APRM Downscale	≥ 5 Indicated on Scale			(9)	6 Instrument Channels	A or B
2	APRM High Flux in Startup	≤ 15% Power	X	X		6 Instrument Channels	A
2	High Reactor Pressure	≤ 1035 psig	X(8)	X	X	4 Instrument Channels	A

3.1 BASES

The reactor protection system automatically initiates a reactor scram to:

1. Preserve the integrity of the fuel cladding.

2. Preserve the integrity of the reactor coolant system.

3. Minimize the energy which must be absorbed following a loss-of-coolant accident, and prevent inadvertant criticality.

This specification provides the limiting conditions for operation necessary to preserve the ability of the system to perform its intended function even during periods when instrument channels may be out of service because of maintenance. When necessary, one channel may be made inoperable for brief intervals to conduct required functional tests and calibrations.

The reactor protection system is of the dual channel type (Reference Subsection 7.2 of the Updated FSAR). The system is made up of two independent trip systems, each having three subchannels of tripping devices. One of the three subchannels has inputs

The APRM (High flux in Startup or Refuel) system provides protection against excessive power levels and short reactor periods in the startup and intermediate power ranges.

The IRM system provides protection against short reactor periods in these ranges.

A source range monitor (SRM) system is also provided to supply additional neutron level information during startup but has no scram functions (reference paragraph 7.6.1.4 of the Updated FSAR). Thus, the IRM and APRM are required in the "Refuel" and "Startup/Hot Standby" modes. In the power range the APRM system provides required protection (reference paragraph 7.6.1.7 of the Updated FSAR). Thus the IRM System is not required in the "Run" mode. The APRM's cover only the power range. The IRM's and APRM's provide adquate coverage in the startup and intermediate range.

The control rod drive scram system is designed so that all of the water which is discharged from the reactor by a scram can be accommodated in the discharge piping. The scram discharge volume accommodates in excess of 60 gallons of water and is the low point in the piping. No credit was taken for this volume in the design of the discharge piping as concerns the amount of water which must be accommodated during a scram.

to the Refuel mode during reactor power operation does not diminish the protection provided by the reactor protection system.

Turbine stop valve closure trip occurs at approximately 10% of valve closure. Below approximately 192 psig turbine first stage pressure (30% of rated power), the scram signal due to turbine stop valve closure is by-passed because the flux and pressure scrams are adequate to protect the reactor.

Turbine Control valve fast closure scram trip shall initiate within 30 milliseconds of the start of control valve fast closure. The trip level setting is verified by measuring the time interval from energizing the fast acting solenoid (from valve test switch) to pressure switch response; the measured result is compared to base line data taken during each refueling outage. Turbine control valve fast closure is sensed by measuring disc dump electro-hydraulic oil line pressure (Relay Emergency Trip Supply) which decreases rapidly upon generator load rejection. This scram is only effective when turbine steam flow is above 30% of rated as measured by turbine first stage pressure (approximately 206 psia).

The requirement that the IRM's be inserted in the core when the APRM's read < 5 as indicated on the scale in the Startup and Refuel modes assures that there is proper overlap in the neutron monitoring system functions and thus, that adequate coverage is provided for all ranges of reactor operation.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
C. Control Rod Block Actuation	C. Control Rod Block Actuation
1. The limiting conditions of operation for the instrumentation that initiates control rod block are given in Table 3.2-C.	Instrumentation shall be functionally tested, calibrated and checked as indicated in Table 4.2-C.
2. The minimum number of operable instrument channels specified in Table 3.2-C for the Rod Block Monitor may be reduced by one in one of the trip systems for maintenance and/or testing, provided that this condition does not last longer than a total of 24 hours in any thirty day period.	System logic shall be functionally tested as indicated in Table 4.2-C.
D. Radiation Monitoring Systems - Isolation & Initiation Functions	D. Radiation Monitoring Systems - Isolation & Initiation Functions
1. Steam Air Ejector Offgas System	1. Steam Air Ejector Offgas System
(a) Except as specified in (b) below, both post treatment steam air ejector offgas system radiation monitors shall be operable during reactor power operation. The trip settings for the monitors shall be set at a value not to exceed the equivalent of the stack release limit specified in the Environmental Technical Specifications. The Steam Air Ejector isolation valves close immediately if the Steam Air Ejector Offgas Radiation Monitor output exceeds the trip setting.	Instrumentation shall be functionally tested, calibrated and checked as indicated in Table 4.2.D. System logic shall be functionally tested as indicated in Table 4.2-D.
(b) From and after the date that one of the two steam air ejector radiation monitors is made or found to	

TABLE 3.2-A

INSTRUMENTATION THAT INITIATES PRIMARY CONTAINMENT ISOLATION (continued)

Minimum No. of Operable Instrument Channels Per Trip System (1)	Instrument	Trip Level Setting	Number of Instrument Channels Provided by Design	Valve Groups Operated by Signal	Action (2)
1	Reactor Cleanup Area Ambient High Temperature	$\leq$ 130°F	3	5	D
1	Reactor Cleanup Area Differential High Temperature	Δ 14°F*	3	5	D
2	Loss of Main Condensor Vacuum	$\leq$ 10 in Hg Vacuum	4	1	B
2	Reactor Low-Low Water Level	At or above +119.5" indicated level (3)	4	8	A

*Note: The actual setpoint shall be Δ14°F above the 100% operation ambient temperature conditions as determined by DAEC Plant Test Procedure.

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TABLE [illegible] (Continued)

INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

Minimum No. of Operable Instrument Channels Per Trip System (1)	Trip Function	Trip Level Setting	Number of Instrument Channels Provided by Design	Remarks
1	Reactor Low Level (inside shroud)	$\geq$ + 305.5 in. above vessel zero (2/3 core height or $\geq$ -39 in. TAF)	2 Instrument Channels	Prevents inadvertent operation of containment spray during accident condition
2	Containment High Pressure	1 < p < 2 psig	4 Instrument Channels	Prevents inadvertent operation of containment spray during accident condition
1	Confirmatory Low Level	$\leq$ + 170 in. indicated level (4)	2 Instrument Channels	ADS Permissive
2	High Drywell Pressure	$\leq$ 2.0 psig	4 HPCI Instrument Channels	1. Initiates Core Spray LPCI; HPCI
2	Reactor Low Pressure	$\geq$ 450 psig	4 Instrument Channels	Permissive for open Core Spray and LCPI Injection valves. Coincident with high drywell pressure, start LPCI and Core Spray pumps

TABLE 3.2 (Continued)

INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

Minimum No. of Operable Instrument Channels Per Trip System (1)	Trip Function	Trip Level Setting	Number of Instrument Channels Provided by Design	Remarks
1	RCIC Leak Detection Time Delay	$\leq$ 30 min.	2 Instrument Channels	
2 (5)	HPCI Steam Line Low Pressure	100 > P > 50 psig (3)	4 Instrument Channels	
2	HPCI Equipment Room High Ambient Temperature	$\leq$ 175°F	4 Instrument Channels	2 Channels for auto isolation and 2 channels for annunciation
2	HPCI Equipment Room High Differential Temperature	$\leq \Delta$ 50°F (3)	4 Instrument Channels	
1 per 4 KV Bus	4 KV Emergency Bus Undervoltage	20 $\leq$ V $\leq$ 28 Volts	2	1. Trips all loaded breakers 2. Fast transfer permissive 3. Dead bus start of diesel
1 per 4kV Bus	4 kV Emergency Bus Sequential Loading Relay	$\geq$ 65% of Rated Voltage	2	Permits sequencing of vital loads
2 per 4kV Bus	Emergency Transformer Undervoltage	$\geq$ 65% of Rated Voltage	4	1. Trips emergency transformer feed to 4kV emergency bus 2. Fast transfer permissive
1 per 4kV Bus (7)	4kV Emergency Bus Degraded Voltage	108 < V < 111 volts 8.0 $\leq$ T.D. $\leq$ 8.5 sec.	2 Matrices	1. Trips 4kV emergency bus incoming breakers 2. Starts diesel 3. Permits sequence of vital loads

TABLE 3.2-C

INSTRUMENTATION THAT INITIATES CONTROL ROD BLOCKS

Minimum No. of Operable Instrument Channels Per Trip System (1)	Trip Function	Trip Level Setting	Number of Instrument Channels Provided by Design	Action
2	APRM Upscale (Flow Biased)	$\leq (0.66\ W + 42)\left(\frac{FRP}{MFLPD}\right)^{(2)}$	6 inst. Channels	(1)
2	APRM Upscale (Not in Run Mode)	$\leq$ 12 indicated on scale	6 Inst. Channels	(1)
2	APRM Downscale	$\geq$ 5 indicated on scale	6 Inst. Channels	(1)
1 (7)	Rod Block Monitor (Flow Biased)	$\leq (0.66\ W + 39)\left(\frac{FRP}{MFLPD}\right)^{(2)}$	6 Inst. Channels	(1)
1 (7)	Rod Block Monitor Downscale	$\geq$ 5 indicated on scale	2 Inst. Channels	(1)
2	IRM Downscale (3)	$\geq$ 5/125 full scale	6 Inst. Channels	(1)
2	IRM Detector not in Startup Position	(8)	6 Inst. Channels	(1)
2	IRM Upscale	$\leq$ 108/125	6 Inst. Channels	(1)
2 (5)	SRM Detector not in Startup Position	(4)	4 Inst. Channels	(1)
2 (5)(6)	SRM Upscale	$\leq 10^5$ counts/sec.	4 Inst. Channels	(1)
1	Scram Discharge Volume Water Level High	$\leq$ 24 gallons	1 Inst. Channel	(9)

explicitly stated where the high and low values are both critical and may have a substantial effect on safety. The setpoints of other instrumentation, where only the high or low end of the setting has a direct bearing on safety, are chosen at a level away from the normal operating range to prevent inadvertent actuation of the safety system involved and exposure to abnormal situations.

Actuation of primary containment valves is initiated by protective instrumentation shown in Table 3.2-A which senses the conditions for which isolation is required. Such instrumentation must be available whenever primary containment integrity is required.

The instrumentation which initiates primary system isolation is connected in a dual bus arrangement.

Many of the reactor water level trip settings are defined or described in terms of "inches above the top of the active fuel." In the new reload fuel the column of fuel pellets in each fuel pin of a bundle is 150 inches long; whereas in the initial core load and first few reloads it was 144 inches long. Thus, during the period of reloads until all of the 144 inch bundles are replaced with bundles with 150 inches of fuel pellets the core will be composed of fuel bundles with fuel pins containing differing lengths of fuel pellet columns and the term "top of active fuel" no longer has a precise physical meaning. Since the basis of all safety analyses is the absolute level (inches above vessel zero) of the trip settings, the "top of the active fuel" has been arbitrarily defined to be 344.5 inches above vessel zero. This definition is the same as that given by Figure 5.1-1 of the Updated FSAR for the initial core and maintains the consistency between the various level definitions given in the FSAR and the technical specifications.

adequate to prevent uncovering the core in the case of a break in the largest line assuming a 60 second valve closing time. Required closing times are less than this.

The low-low reactor water level instrumentation is set to trip when reactor water level is 119.5" above top of the active fuel. This trip initiates the HPCI and RCIC and trips the recirculation pumps. The low-low-low reactor water level instrumentation is set to trip when the water level is 18.5" above the top of the active fuel. This trip activates the remainder of the CSCS subsystems, closes Group 7 valves, closes Main Steam Line Isolation Valves, Main Steam Drain Valves, Recirc Sample Valves (Group 1) and starts the emergency diesel generators. These trip level settings were chosen to be high enough to prevent spurious actuation but low enough to initiate CSCS operation and primary system isolation so that post accident cooling can be accomplished and the guidelines of 10 CFR 100 will not be exceeded. For large breaks up to the complete circumferential break of a 22-inch recirculation line and with the trip setting given above, CSCS initiation and primary system isolation are initiated in time to meet the above criteria. Reference Sections 6.3 and 7.3 of the Updated FSAR.

The high drywell pressure instrumentation is a diverse signal for malfunctions to the water level instrumentation and in addition to initiating CSCS, it causes isolation of Group 2 and 3 isolation valves. For the breaks discussed above, this instrumentation will generally initiate CSCS operation before the low-low-low water level instrumentation; thus the results given above are applicable here also. See Specification 3.7 for Isolation Valve Closure Group. The water level instrumentation initiates protection for the full spectrum of loss-of-coolant accidents and causes isolation of all isolation valves except Group 6.

Venturis are provided in the main steam lines as a means of measuring steam flow and also limiting the loss of mass inventory from the vessel during a steam line break accident. The primary function of the instrumentation is to detect a break in the main steam line. For the worst case accident, main steam line break outside the drywell, a trip setting of 140% of rated steam flow in conjunction with the flow limiters and consequently main steam line valve closure, limits the mass inventory loss such that fuel is not uncovered, fuel clad temperatures peak at approximately 1000°F and release of radioactivity to the environs is below 10 CFR 100 guidelines. Reference Subsection 15.6.5 of the Updated FSAR.

Temperature monitoring instrumentation is provided in the main steam line tunnel and turbine building to detect leaks in this area. Trips are provided on this instrumentation and when exceeded, cause closure of isolation valves. See Spec 3.7 for Valve Group. The setting is 200°F for the main steam line tunnel detector. For large breaks, the high steam flow instrumentation is a backup to the temperature instrumentation.

High radiation monitors in the main steam line tunnel have been provided to detect gross fuel failure as in the control rod drop accident. With the established setting of 3 times normal background, and main steam line isolation valve closure, fission product release is limited so that 10 CFR 100 guidelines are not exceeded for this accident. Reference Subsection 15.4.7 of the Updated FSAR.

Pressure instrumentation is provided to close the main steam isolation valves in RUN Mode when the main steam line pressure drops below 850 psig. The Reactor Pressure Vessel thermal transient due to an inadvertent opening of the turbine bypass valves when not in the RUN Mode is less severe than the loss of feedwater analyzed in Subsection 15.6.3 of the Updated FSAR, therefore, closure of the Main Steam Isolation valves for thermal transient protection when not in RUN Mode is not required.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
e. If Specification 3.3.B.3a through d cannot be met, the reactor shall not be started, or if the reactor is in the run or startup modes at less than 30% rated power, it shall be brought to a shutdown condition immediately.	1) The correctness of the control rod withdrawal sequence input to the RWM computer shall be verified. 2) The RWM computer on line diagnostic test shall be successfully performed.
f. The sequence restraints imposed on the control rods may be removed by the use of the individual rod position bypass switches for scram testing only those rods which are fully withdrawn in the 100% to 50% rod density range.	3) Proper annunciation of the selection error of at least one out-of-sequence control rod in each fully inserted group shall be verified. 4) The rod block function of the RWM shall be verified by withdrawing the first rod as an out-of-sequence control rod no more than to the block point. c. When required, the presence of a second licensed operator to verify the following of the correct rod program shall be verified.
4. Control rods shall not be withdrawn for startup or refueling unless at least two source range channels have an observed count rate equal to or greater than three counts per second.	4. Prior to control rod withdrawal for startup or during refueling, verify that at least two source range channels have an observed count rate of at least three counts per second.
5. During operation with limiting control rod patterns, either: a. Both RBM channels shall be operable: or b. Control rod withdrawal shall be blocked: or c. The operating power level shall be limited so that the MCPR will remain above safety limit assuming a single error that results in complete withdrawal of any single operable control rod.	5. When a limiting control rod pattern exists, an instrument functional test of the RBM shall be performed prior to withdrawal of the designated rod(s).

LIMITING CONDITION FOR OPERATION

C. Scram Insertion Times

1. The average scram insertion time, based on the de-energization of the scram pilot valve at time zero, of all operable control rods in the reactor power operation condition shall be no greater than:

Rod Position	Average Scram Insertion Times (Sec)
46	0.361
36	0.917
26	1.479
06	2.693

2. The average scram insertion times for the three fastest control rods of all groups of four control rods in a 2 x 2 array shall be no greater than:

Rod Position	Average Scram Insertion Times (Sec)
46	0.383
36	0.972
26	1.556
06	2.847

3. Maximum scram insertion time for 90% insertion of any operable control rod should not exceed 7.00 seconds.

SURVEILLANCE REQUIREMENT

C. Scram Insertion Times

1. After each refueling outage all operable rods shall be scram time tested from the fully withdrawn position with the nuclear system pressure above 950 psig (with saturation temperature) and the requirements of Specification 3.3.B.3.a met. This testing shall be completed prior to exceeding 40% power. Below 30% power, only rods in those sequences (A_{12} and A_{34} or B_{12} and B_{34}) which were fully withdrawn in the region from 100% rod density to 50% rod density shall be scram time tested. During all scram time testing below 30% power, the Rod Worth Minimizer shall be operable or a second licensed operator shall verify that the operator at the reactor console is following the control rod program.

3.3 and 4.3 BASES

1. Reactivity Limitation

   a. The requirements for the control rod drive system have been identified by evaluating the need for reactivity control via control rod movement over the full spectrum of plant conditions and events. As discussed in Subsection 4.6.1 of the Updated FSAR, the control rod system design is intended to provide sufficient control of core reactivity that the core could be made subcritical with the strongest rod fully withdrawn. This reactivity characteristic has been a basic assumption in the analysis of plant performance. Compliance with this requirement can be demonstrated conveniently only at the time of initial fuel loading or refueling. Therefore, the demonstration must be such that it will apply to the entire subsequent fuel cycle. The demonstration shall be performed with the reactor core in the cold, xenon-free condition and will show that the reactor is subcritical by at least R + 0.38% Δ k/k with the analytically determined strongest control rod fully withdrawn.

2. Control Rod Withdrawal

a. Control rod drop accidents as discussed in the FSAR can lead to significant core damage. If coupling integrity is maintained, the possibility of a rod drop accident is eliminated. The overtravel position feature provides a positive check as only uncoupled drives may reach this position. Neutron instrumentation response to rod movement provides a verification that the rod is following its drive. Absence of such response to drive movement could indicate an uncoupled condition. Rod position indication is required for proper function of the rod sequence control system and the rod worth minimizer (RWM).

b. 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 Subsection 4.6.1 of the Updated FSAR and the safety evaluation is given in Subsection 4.6.2 of the Updated FSAR. 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. Additionally, the support is not required if all control rods are fully inserted and if an adequate shutdown margin with one control rod withdrawn has been demonstrated, since the reactor would remain subcritical even in the event of complete ejection of the strongest control rod.

c. The Rod Worth Minimizer (RWM) and the Rod Sequence Control System (RSCS) restrict withdrawals and insertions of control rods to prespecified sequences. These sequences are established such that the drop of any in-sequence control rod from the fully inserted position to the position of the control rod drive would not cause the reactor to sustain a power excursion resulting in a peak fuel enthalpy in excess of 280 cal/gm. An enthalpy of 280 cal./gm. is well below the level at which rapid fuel dispersal could occur (i.e., 425 cal./gm.).

Primary system damage in this accident is not possible unless a significant amount of fuel is rapidly dispersed. Ref. Subsections 4.3.1, 7.7.4.9, and 15.4.7 of the Updated FSAR.

In performing the function described above, the RWM and RSCS are not required to impose any restrictions at core power levels in excess of 20% of rated. Material in the cited references shows that it is impossible to reach 280 calories per gram in the event of a control rod drop occurring at power greater than 20%, regardless of the rod pattern. This is true for all normal and abnormal patterns including those which maximize the individual control rod worth.

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 written sequences. 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.

A limiting control rod pattern is a pattern which results in the core being on a thermal hydraulic limit. During use of such patterns, it is judged that testing of the RBM system prior to withdrawal of such rods to assure its operability will assure that improper withdrawal does not occur. It is the responsibility of the Reactor Engineer to identify these limiting patterns and the designated rods either when the patterns are initially established or as they develop due to the occurrence of inoperable control rods in other than limiting patterns. Other personnel qualified to perform this function may be designated by the Plant Superintendent-Nuclear.

3. Scram Insertion Times

The control rod system is designed 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 safety limit.

4. Reactivity Anomalies

During each fuel cycle excess operative 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 at 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%ΔK. Deviations in core reactivity greater than 1% ΔK are not expected and require thorough evaluation. One percent 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.

4.4 BASES

Standby Liquid Control System

Experience with pump operability indicates that the monthly test, in combination with the tests during each operating cycle, is sufficient to maintain pump performance. The only practical time to fully test the liquid control system is during a refueling outage. Various components of the system are individually tested periodically, thus making unnecessary more frequent testing of the entire system.

The details of the various tests are discussed in the Updated FSAR Subsection 9.3.4. The solution temperature and volume are checked at a frequency to assure a high reliability of operation of the system should it ever be required.

3.5 BASES

A. Core Spray and LPCI Subsystems

This specification assures that adequate emergency cooling capability is available whenever irradiated fuel is in the reactor vessel.

Based on the loss-of-coolant accident (LOCA) evaluation models described in General Electric Topical Report NEDO-21082-02-1A(1977) and the performance evaluation given in Subsection 6.3.3 of the Updated FSAR and in accordance with 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors," Either of the two core spray subsystems and the LPCI subsystem provides sufficient cooling to the core to dissipate the energy associated with the loss-of-coolant accident, to limit calculated fuel clad temperature to less than 2200°F to assure that core geometry remains intact, and to limit clad metal-water reaction to less than 1%.

The limiting conditions of operation in Specification 3.5.A.1 through 3.5.A.6 specify the combinations of operable subsystems to assure the availability of the minimum cooling systems noted above.

flow and the liquid inventory begins to rise. This type of response is typical of the small breaks. The core never uncovers and is continously cooled throughout the transient so that no core damage of any kind occurs for breaks that lie within the capacity range of the HPCI.

As mentioned in Section 6.3.1 of the Updated FSAR, the ADS provides a single failure proof path for depressurization for postulated transients and accidents. The RCIC serves as an alternate to the HPCI only for decay heat removal even if the feedwater is assumed to be lost. Considering the HPCI and the ADS plus RCIC as redundant paths, reference (1) methods would give an estimated allowable repair time of 15 days based on the one month testing frequency. However, a maximum allowable repair time of 7 days is selected for conservatism.

The HPCI and RCIC as well as all other Core Standby Cooling Systems must be operable when starting up from a Cold Condition. It is realized that the HPCI is not designed to operate until reactor pressure exceeds 150 psig and is automatically isolated before the reactor pressure decreases below 100 psig. It is the intent of this specification to assure that when the reactor is being started up from a Cold Condition, the HPCI is not known to be inoperable.

E. RCIC System

The RCIC is designed to provide makeup to the nuclear system as part of the planned operation for periods when the main condenser is unavailable. RCIC also serves for decay heat removal when feedwater is lost. In all other postulated accidents and transients, the ADS provides redundancy for the HPCI. Based on this and judgements on the reliability of the HPCI system, an allowable repair time of 1 month is justified, however, a maximum allowable repair time of 7 days is selected for conservatism. Immediate and weekly demonstrations of HPCI operability during RCIC outage is considered adequate based on judgement and practicality.

F. Automatic Depressurization System (ADS)

The operability of the ADS under all conditions of depressurization of the nuclear system automatically or manually, insures an essential response to station abnormalities.

The nuclear system pressure relief system provides automatic nuclear system depressurization for small breaks in the nuclear system so that the low pressure coolant injection (LPCI) and the core spray subsystems can operate to protect the fuel barrier.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
3. The reactor vessel head bolting studs shall not be under tension unless the temperature of the vessel head flange and the head is greater than 100°F.	Test specimens of the reactor vessel base, weld and heat affected zone metal subjected to the highest fluence of greater than 1 MeV neutrons shall be installed in the reactor vessel adjacent to the vessel wall at the core midplane level. The specimens and sample program shall conform to ASTM E 185-66 or ASTM E-185-70 to the degree discussed in Section 5.3.1.6 of the Updated FSAR. Samples shall be withdrawn at one-fourth and three-fourths service life in accordance with 10 CFR 50, Appendix H.
4. The pump in an idle recirculation loop shall not be started unless the temperatures of the coolant within the idle and operating recirculation loops are within 50°F of each other.	
5. The reactor recirculation pumps shall not be started unless the coolant temperatures between the dome and the bottom head drain are within 145°F.	
	3. When the reactor vessel head bolting studs are tensioned and the reactor is in a Cold Condition, the reactor vessel shell temperature immediately below the head flange shall be permanently recorded.
	4. Prior to and during startup of an idle recirculation loop, the temperature of the reactor coolant in the operating and idle loops shall be permanently logged.
	5. Prior to starting a recirculation pump, the reactor coolant temperatures in the dome and in the bottom head drain shall be compared and permanently logged.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
	h. Whenever the I-131 dose equivalent as determined in (e) above exceeds 0.6 μCi/gm (50% of the equilibrium value) notify the Regional Administrator, Region III, in writing within 10 days.
2.a The reactor coolant water shall not exceed the following limits with steaming rates less than 100,000 pounds per hour, during shutdown or when in the refueling mode: Conductivity 5 μmho/cm Chloride ion 0.1 ppm	2. A sample of reactor coolant shall be analyzed: a. At least every 4 hours during startup and at steaming rates below 100,000 pounds per hour for chloride ion content if the conductivity is above 0.5 μmho/cm or if it increases at a rate of 0.2 μmho/cm/hr or more. The minimum frequency will be once per day.
At all times when the conductivity exceeds 5 micromhos/cm, the pH shall not be less than 4.6, except that short-term spikes of up to two hours duration each are permissible in the pH range 4.0 to 4.5 and of up to four hours duration each, in the range 4.5 to 4.6. The total time in which the conductivity exceeds 5 micromhos/cm shall not exceed 720 hours.	

3.6.A and 4.6.A BASES:

Thermal and Pressurization Limitations

The thermal limitations for the reactor vessel meet the requirements of 10 CFR 50, Appendix G.

The allowable rate of heatup and cooldown for the reactor vessel contained fluid is 100°F per hour averaged over a period of one hour. This rate has been chosen based on past experience with operating power plants. The associated time period for heatup and cooldown cycles when the 100°F per hour rate is limiting provides for efficient, but safe, plant operation.

Specific analyses were made based on a heating and cooling rate of 100°F/hour applied continuously over a temperature range of 100°F to 546°F. Calculated stresses were within ASME Boiler and Pressure Vessel Code Section III stress intensity and fatigue limits even at the flange area where maximum stress occurs.

Chicago Bridge and Iron Company performed detailed stress analysis as shown in the Updated FSAR Appendix 5A, "Site Assembly of the Reactor Vessel." This analysis includes more severe thermal conditions than those which would be encountered during normal heating and cooling operations.

The permissible flange to adjacent shell temperature differential of 145°F is the maximum calculated for 100°F hour heating and cooling rate applied continuously over a 100°F to

3.6.D & 4.6.D BASES:

Safety and Relief Valves

The pressure relief system has been sized to meet two design bases. First, the total safety/relief valve capacity has been established to meet the overpressure protection criteria of the ASME Code. Second, the distribution of this required capacity between safety valves and relief valves has been set to meet power generation design basis #1 of Section 5.4.13.1 of the Updated FSAR, which states that the nuclear system relief valves shall prevent opening of the safety valves during normal plant isolations and load rejections.

The details of the analysis which shows compliance with the ASME Code requirements is presented in Subsection 5.4.13 of the Updated FSAR and is reverified in individual reload analyses.

Six relief valves and two safety valves are installed. The analysis of the worst overpressure transient, (3-second closure of all main steam line isolation valves) neglecting

3.6.F & 4.6.F BASES:

Jet Pump Flow Mismatch

The LPCI loop selection logic has been previously described in the Updated FSAR Section 7.3.1.1.24. For some limited low probability accidents with the recirculation loop operating with large speed differences, it is possible for the logic to select the wrong loop for injection. For these limited conditions the core spray itself is adequate to prevent fuel temperatures from exceeding allowable limits. However, to limit the probability even further, a procedural limitation has been placed on the allowable variation in speed between the recirculation pumps.

The licensee's analyses indicate that above 80% power the loop select logic could be expected to function at a speed differential up to 14% of their average speed. Below 80% power the loop select logic would be expected to function at a speed differential up to 20% of their average speed. This specification provides margin because the limits are set at ±10% and ±15% of the average speed for the above and below

The type of inspection planned for each component depends on location, accessibility, and type of expected defect. Direct visual examination is proposed wherever possible since it is fast and reliable. Surface inspections are planned where practical, and where added sensitivity is required. Ultrasonic testing or radiography shall be used where defects can occur in concealed surfaces. Section 5.2.4 of the DAEC Updated FSAR provides details of the inservice inspection and testing program.

3.6.H & 4.6.H BASES:

Shock Suppressors (Snubbers)

Snubbers are designed to prevent unrestrained pipe motion under dynamic loads as might occur during an earthquake or severe transient, while allowing normal thermal motion during startup and shutdown. The consequence of an inoperable snubber is an increase in the probability of structural damage to piping as a result of a seismic or other event initiating dynamic loads. It is therefore required that all snubbers required to protect the primary coolant system or any other safety system or component be operable during reactor operation.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
b. One drywell-suppression chamber vacuum breaker may be non-fully closed as indicated by its position lights so long as it is determined that total drywell to suppression pool bypass area of less than 0.2 square feet exists.	b. When a vacuum breaker indicates non-fully closed, the continuous leak rate monitoring system shall be utilized immediately and every 30 days thereafter to determine that a bypass area of not more than 0.2 square feet exists. A detailed description of allowable drywell to suppression pool bypass leakage is found in Section 6.2.1.3.5 of the Updated FSAR.
c. One drywell-suppression chamber vacuum breaker may be determined to be inoperable for opening.	c. When it is determined that a vacuum breaker valve is inoperable for opening at a time when operability is required, all other vacuum breaker valves shall be exercised immediately and every 15 days thereafter until the inoperable valve has been returned to normal service. Once each operating cycle, each vacuum breaker valve shall be visually inspected to insure proper maintenance and operation.
d. If specifications 3.7.A.4.a, .b, or .c cannot be met, the situation shall be corrected within 24 hours or the unit will be placed in a cold shutdown condition in an orderly manner.	d. A leak test of the drywell to suppression chamber structure shall be conducted at each refueling. The drywell pressure will be increased by approximately 1 psi with respect to the wetwell pressure and held constant. The 2 psig scram setpoint will not be exceeded. The subsequent wetwell pressure transient (if any) will be monitored with a precision pressure gauge capable of detecting a small pressure increase. If the drywell pressure cannot be increased by 1 psi over the wetwell pressure, it would be because excess leakage exists. Excess leakage

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
	would require the leakage source to be identified and eliminated before primary system pressurization. A more detailed description of this test is found in Section 6.2.6.3.5.3 of the Updated FSAR.
5. Oxygen Concentration	5. Oxygen Concentration
a. After completion of the startup test program and demonstration of plant electrical output, the primary containment atmosphere shall be reduced to less than 4% oxygen with nitrogen gas during reactor power operation with reactor coolant pressure above 90 psig, except as specified in 3.7.A.5.b.	The primary containment oxygen concentration shall be measured and recorded at least twice weekly.
b. Within the 24-hour period subsequent to placing the reactor in the Run mode with the intent of commencing power operation following a shutdown, the containment atmosphere oxygen concentration shall be reduced to less than 4% by volume and maintained in this condition. Deinerting may commence 24 hours prior to a shutdown.	

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
6. Containment Atmosphere Dilution	6. Containment Atmosphere Dilution
a. Whenever the reactor is in power operation, the Post-LOCA Containment Atmosphere Dilution System must be operable and capable of supplying nitrogen to the containment for atmosphere dilution if required by post-LOCA conditions. If this specification cannot be met, the system must be restored to an operable condition within 7 days.	a. The post-LOCA containment atmosphere dilution system shall be functionally tested once per operating cycle.
b. Whenever the reactor is in power operation, the post-LOCA Containment Atmosphere Dilution System shall contain a minimum of 50,000 scf of N_2 as determined by pressure and temperature measurements. If this specification cannot be met, the minimum volume will be restored within 7 days.	b. The volume in the N_2 storage bank shall be recorded weekly.
c. Whenever the reactor is in power operation, there shall be at least one CAD system H_2 and O_2 analyzer serving the drywell and the suppression chamber. If this specification cannot be met, the reactor must be taken out of power operation.	c. The CAD system H_2 and O_2 analyzers shall be tested for operability using standard bottled H_2 and O_2 once per month and shall be calibrated once per 6 months. The atmosphere analyzing system shall be functionally tested once per operating cycle in conjunction with specification 4.7.A.6.a. Should one of the two H_2 or O_2
d. If Specifications 3.7.A.6a and b cannot be met, the reactor shall be brought to at least the hot shutdown condition within 24 hours.	

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
	functionally tested once per operating cycle in conjunction with specification 4.7.A.6.a. Should one of the two H_2 or O_2 analyzers serving the drywell or suppression pool be found inoperable, the remaining analyzer of the same type serving the same compartment shall be tested for operability once per week until the defective analyzer is made operable.
7. Drywell-Suppression Chamber Differential Pressure	7. Drywell-Suppression Chamber Differential Pressure
a. Differential pressure between the drywell and suppression chamber shall be maintained at equal to or greater than 1.10 psid except as specified in (1) and (2) below:	a. The pressure differential between the drywell and suppression chamber shall be recorded at least once each shift.
(1) Within the 24-hour period subsequent to placing the reactor in the Run Mode following a shutdown, the differential shall be established. The differential may be decreased to less than 1.10 psid 24 hours prior to a scheduled shutdown.	
(2) This differential may be decreased to less than 1.10 psid for a maximum of four hours during required operability testing of the HPCI system pump, the RCIC system pump, the drywell-pressure suppression chamber vacuum breakers, the suppression chamber to reactor building vacuum breakers, and to perform leak rate testing required by specification 4.7.A.2.d.4, and to allow for inerting operations to satisfy specification 3.7.A.5 requirements.	
b. If the differential pressure of specification 3.7.A.7.a cannot be maintained, and the differential pressure cannot be restored within the subsequent six (6) hour period, an orderly shutdown shall be initiated and the reactor shall be in the Cold Shutdown condition within the following 24 hours.	

Using a 50°F rise (Table 6.2-1, Updated FSAR) in the suppression chamber water temperature and a minimum water volume of 58,900 ft^3, the 170°F temperature which is used for complete condensation would be approached only if the suppression pool temperature is 120°F prior to the DBA-LOCA. Maintaining a pool temperature of 95°F will assure that the 170°F limit is not approached.

2. Inerting

Safety Guide No. 7 assumptions for metal-water reactions result in hydrogen concentrations in excess of the Safety approximately once per year, assures the paint is intact. Experience with this type of paint at fossil fueled generating stations indicates that the inspection interval is adequate.

6. Post LOCA Atmosphere Dilution

In order to ensure that the containment atmosphere remains inerted, i.e., the oxygen-hydrogen mixture below the flammable limit, the capability to inject nitrogen into the containment after a LOCA is provided. The CAD system serves as the post-LOCA Containment Atmosphere Dilution System. By maintaining a minimum of 50,000 scf of liquid N_2 in the storage bank it is assured that a seven-day supply of N_2 for post-LOCA containment inerting is available.

The Post-LOCA Containment Atmosphere Dilution System design basis and description are presented in Section 6.2.5 of the Updated FSAR. In summary, the limiting criteria, based on the assumptions of Safety Guide No. 7 are:

1. Maintain oxygen concentration in the containment during post-LOCA conditions to less than 4 Volume %.

the accidents analyzed, as the Updated FSAR Section 15.6.6 for the loss-of-coolant accident shows compliance with 10 CFR 100 guidelines with an assumed efficiency of 95% for the adsorber. Operation of the fans significantly different from the design flow envelope will change the removal efficiency of the HEPA filters and charcoal adsorbers.

Pressure drop across the combined HEPA filters and charcoal adsorbers of less than 11 inches of water at the system design flow rate will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter. Heater capability, pressure drop and air distribution should be determined at least once per operating cycle to show system performance capability.

The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. Tests of the charcoal adsorbers with halogenated hydrocarbon refrigerant shall be performed in accordance with USAEC Report DP-1082. Iodine removal efficiency tests shall follow RDT Standard M-16-1T. (The design of the SGTS system allows the removal of charcoal samples from the bed directly through the use of a grain thief.) Each sample should be at least two inches in diameter and a length equal to the thickness of the bed. If test results are unacceptable, all adsorbent in the system shall be replaced with an adsorbent qualified according

3.7.A & 4.7.A REFERENCES

1. Section 6.2 of the Updated FSAR.

2. ASME Boiler and Pressure Vessel Code, Nuclear Vessels, Section III, maximum allowable internal pressure is 62 psig.

3. Staff Safety Evaluation of DAEC, USAEC, Directorate of Licensing, January 23, 1973.

4. 10 CFR 50.54, Appendix J, Reactor Containment Testing Requirements, Federal Register, August 27, 1971.

5. DAEC Short-Term Program Plant Unique Analysis, NUTECH Doc. No. IOW-01-065, August 1976.

6. Supplement to DAEC Short-Term Program Plant Unique Analysis, NUTECH Doc. No. IOW-01-071, October 1976.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
4. Reporting Situations covered by Specification 3.8.B.1, 3.8.B.2.b, 3.8.B.2.c, and 3.8.B.3 shall be reported to the NRC within 24 hours, including precautions to be taken and plans for restoration of inoperable components.	

The 250 volt d-c system provides power for the HPCI system. If the battery is taken out of service, the HPCI system would be inoperable and the requirements of Specification 3.5.D this condition must be satisfied.

The 24 volt d-c system provides power for source range monitoring, intermediate range monitoring, and liquid process radiation monitoring. The two neutron monitoring functions are required for safety, however, the design is fail-safe in that loss of 24 volt d-c power would cause the associated trip to function (UFSAR Section 8.3.2).

The battery room is ventilated to prevent accumulation of hydrogen gas exceeding 4 percent concentration. On loss of battery room ventilation, the use of portable ventilation equipment and daily sampling provides assurance that potentially hazardous quantities of hydrogen gas will not accumulate.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT
C. Minimum Critical Power Ratio (MCPR)	C. Minimum Critical Power Ratio (MCPR)
During reactor power operations, MCPR shall be ≥ values as indicated in Table 3.12-2 at rated power and flow. If at any time it is determined by normal surveillance that the limiting value for MCPR is being exceeded, action shall then be initiated within 15 minutes to restore operation to within the prescribed limits. If the operating MCPR is not returned to within the prescribed limits within 2 hours, reduce reactor power to ≤ 25% of Rated Thermal Power within the next 4 hours. Surveillance and corresponding action shall continue until the prescribed limits are again being met.	MCPR shall be determined daily during reactor power operation at ≥ 25% rated thermal power and following any change in power level or distribution that would cause operation with a limiting control rod pattern as described in the bases for Specification 3.3.2. During operation with a limiting control rod pattern, the MCPR shall be determined at least once per 12 hours.
For core flows other than rated the MCPR shall be ≥ values as indicated in Table 3.12-2 times K_f, where K_f is as shown in Figure 3.12-1.	

5.0 DESIGN FEATURES

5.1 SITE

The Duane Arnold Energy Center site is located on the western side of a north-south reach of the Cedar River, approximately 2-1/2 miles north-northeast of the Village of Palo, Iowa. The site consists of approximately 500 acres owned by the Iowa Electric Light and Power Company. The plan of the site is shown on Figures 1.2-1 and 1.2-2 of the Updated FSAR. The minimum distance to the boundary of the exclusion area as defined in 10 CFR 100.3 is approximately 1000 feet.

5.3 REACTOR VESSEL

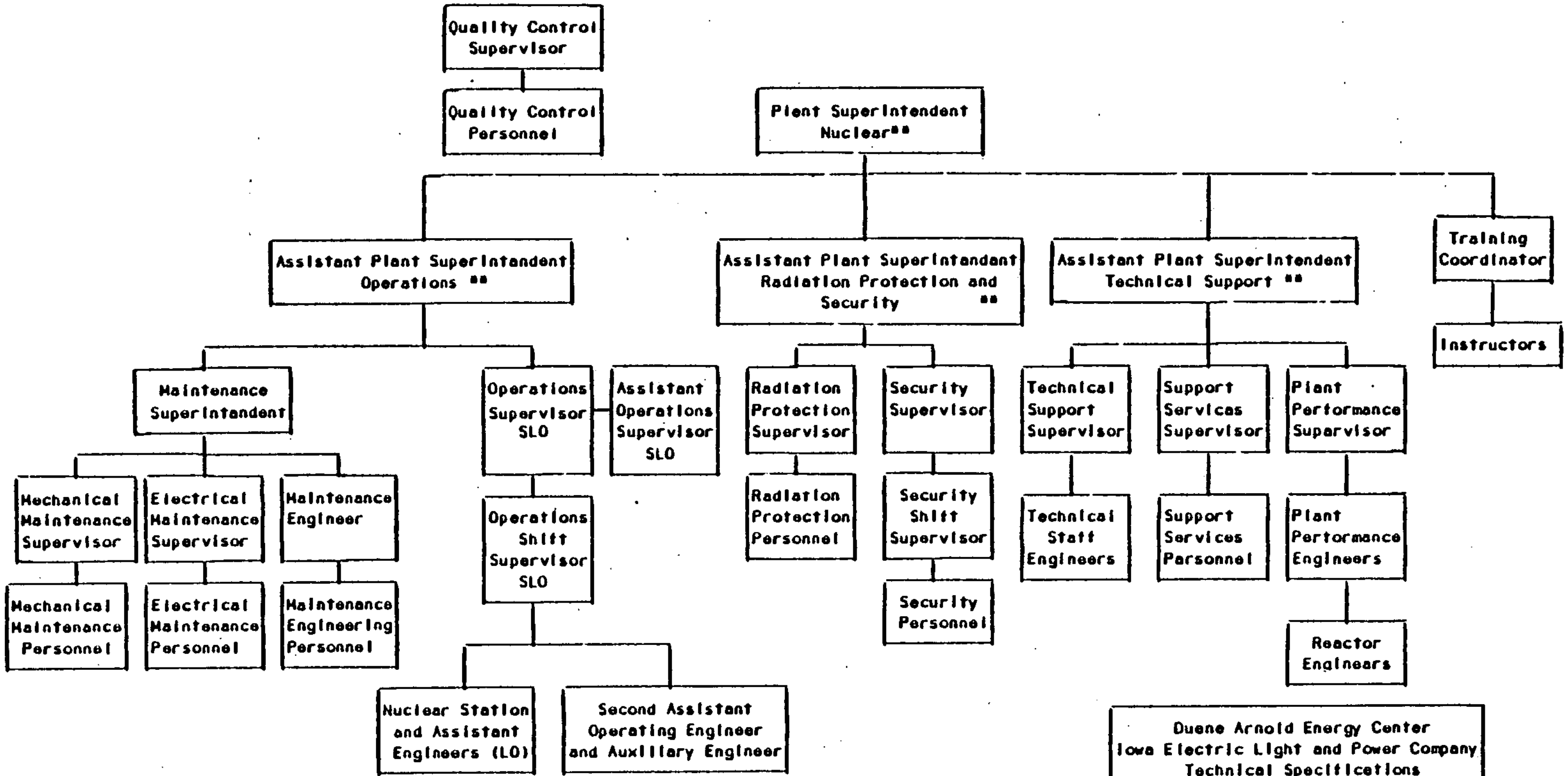
The reactor vessel shall be as described in Table 5.3-5 of the Updated FSAR. The applicable design codes shall be as described in Subsection 5.3.1.1 of the Updated FSAR.

5.4 CONTAINMENT

1. The primary containment and applicable codes shall be as described in Subsection 6.2.1 of the Updated FSAR.

2. The secondary containment shall be as described in Subsection 6.2.3 of the Updated FSAR and the applicable codes shall be as described in Section 6.2.1 of the Updated FSAR.

3. Penetrations to the primary containment and piping passing through such penetrations shall be designed in accordance with standards set forth in Subsection 6.2 of the Updated FSAR.



Duane Arnold Energy Center
Iowa Electric Light and Power Company
Technical Specifications

DAEC Nuclear Plant Staffing
Figure 6.2-1

SLO - Senior Licensed Operator

LO - Licensed Operator

** - Personnel meet ANSI N.18.1-1971 License Requirements

6.7 ACTION TO BE TAKEN IF A SAFETY LIMIT IS EXCEEDED

6.7.1 If a safety limit is exceeded, the reactor shall be shut down and reactor operation shall only be resumed when authorized by the NRC.

6.7.2 An immediate report shall be made to the Director-Nuclear Generation and the Safety Committee. The Director-Nuclear Generation shall promptly report the circumstances to the NRC as specified in Subsection 6.11, Plant Reporting Requirements.

6.7.3 A complete analysis of the circumstances leading up to and resulting from the situation together with recommendations to prevent a recurrence shall be prepared by the Operations Committee. This report shall be submitted to the Director-Nuclear Generation and to the Safety Committee. Appropriate analyses or reports will be submitted to the NRC by the Director-Nuclear Generation as specified in Subsection 6.11, Plant Reporting Requirements.

7. Procedures required by the Emergency Plan.

8. Procedures required by the Plant Security Plan.

9. Operation of radioactive waste systems.

10. Fire Protection Program implementation.

11. A preventative maintenance and periodic visual examination program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient to as low as practical levels. This program shall also include provisions for performance of periodic systems leak tests of each system no less frequently than at refueling cycle intervals.

12. Program to ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions, including training of personnel, procedures for monitoring and provisions for maintenance of sampling and analysis equipment.

6.8.2 Procedures described in 6.8.1 above, and changes thereto, shall be reviewed by the Operations Committee and approved by the Plant Superintendent-Nuclear prior to implementation, except as provided in 6.8.3 below.

6.8.3 Temporary minor changes to procedures described in 6.8.1 above which do not change the intent of the original procedure may be made with the concurrence of two members of the plant management staff, at least one of whom shall hold a senior operator license. Such changes shall be documented and promptly reviewed by the Operations Committee and by the Plant Superintendent-Nuclear. Subsequent incorporation, if necessary, as a permanent change, shall be in accord with 6.8.2 above.

6.8.4 Selected drills of emergency procedures shall be conducted in accordance with the provisions of the Emergency Plan.

6.8.5 The preventive and corrective maintenance program 6.8.1.11 above shall be implemented as follows:

1. Once per year or no less frequently than once per refueling cycle, a detailed walkdown inspection shall be performed and the results recorded.

2. Additional walkdown inspections shall be performed quarterly to detect any visible leakage.

Startup reports shall be submitted with (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial power operation), supplementary reports shall be submitted at least every three months until all three events have been completed.

b. Annual Exposure Report A tabulation on an annual basis of the number of station, utility and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man rem exposure according to work and job functions—/, e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. This tabulation shall be submitted within 60 days of January 1 each year. The dose assignment to various duty functions may be estimates based on pocket dosimeter, TLD, or film badge measurements. Small exposures totaling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific major work functions.

—/ This tabulation supplements the requirements of Sec. 20.407 of 10CFR Part 20.

NOTES TO TABLE 6.11-2

[1]Technical Specifications.

[2]NRC authorization is required prior to performing a change, test, or experiment in this category.

[3]Unique reports covering inspections, tests, and maintenance that are appropriate to assure safe operation of the facility. The frequency and content of these special reports are determined on an individual case basis and designated in the Technical Specifications. Such reports include in-service inspection, tendon surveillance program study, fuel inspection, and containment structural tests.