Docket files

APR 1 9 1974

Docket No. 50-331

Iowa Electric Light and Power Company
ATTN: Duane Arnold, President
Security Building
P. O. Box 351
Cedar Rapids, Iowa 52406

Change No. 1
License No. DPR-49

Gentlemen:

By your letter dated April 17, 1974, you submitted certain proposed changes to the Radiological and Environmental Technical Specifications and Bases for the Duane Arnold Energy Center (DAEC). Our evaluation and action in regard to your proposed changes to the Radiological Technical Specifications and Bases (Appendix A to Operating License DPR-49) are described in this letter. Your proposed changes to the Environmental Technical Specifications and Bases (Appendix B to the Operating License) will be addressed in separate correspondence.

The proposed changes to the Technical Specifications are of minor significance. In general, they involve corrections of typographical errors, word selection errors, inadvertent omissions, and minor inconsistencies in requirements and in certain Bases. No substantive changes to the Technical Specifications and Bases were proposed. Based on our review, we conclude that the proposed changes to the DAEC Radiological Technical Specifications do not present a significant hazards consideration and that there is reasonable assurance that the health and safety of the public will not be endangered by operation of the DAEC in accordance with these changes.

The Regulatory staff's experience with other operating boiling water reactors has indicated a requirement for lower trip settings on main steamline radiation monitors and the addition of an alarm signal, in order to provide maximum information and plant protection without incurring unnecessary reactor scrams and system transients. The trip setting is changed from 6 to 3 times normal background at rated power. A requirement for an alarm signal at 1.5 times normal background at rated power is added. These changes will become effective when the DAEC reaches 50 per cent rated power for the first time.

Pursuant to Section 50.59 of 10 CFR Part 50, the Technical Specifications and Bases for DAEC (Appendix A to Operating License DPR-49) are hereby changed by replacing the present pages 3.1-4, 3.1-7, 3.1-12, 3.1-18, 3.2-5, 3.2-7, 3.2-10, 3.2-12, 3.2-13, 3.2-24, 3.2-31, 3.2-33, 3.2-39, 3.5-1, 3.5-2, 6B

OFFICE➤						
SURNAME➤						
DATE➤						

3.5-5, 3.7-15, 3.7-45, 3.8-2, 3.8-3, 3.8-4, 3.8-11, 3.8-12, 3.9-2, 3.9-3, 3.9-7, 3.10-1, 3.10-2, 3.10-3, 3.10-5, 6.1-1, 6.5-2 and 6.5-5 with the corresponding revised pages, which are enclosed.

These changes to the DAEC Technical Specifications shall become effective on the date of this letter.

Sincerely,

Original signed by
Walter Butler

R. C. DeYoung, Assistant Director
for Light Water Reactors Group 1
Directorate of Licensing

Enclosure: Revised pages to the DAEC Technical Specifications, dated April 1974

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RO (Thornburg) ~~Concurrence~~ concurrence received by phone on 4/19/74 WB

OFFICE➤	L:LWR 1-2	L:LWR 1-2 WB	~~AD:OR~~	~~AD:LWR-1~~ WB		C/P
SURNAME➤	GOwsley:mhp	WRButler	~~KGoller~~	RCDeYoung		
DATE➤	4/19/74	4/19/74	~~4/ /74~~	4/19/74		

TABLE 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) (2.62/P.F.) (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

TABLE 3.1-1 (Continued)

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		
2	High Drywell Pressure	≤ 2.0 psig	X(7)	X(8)	X	4 Instrument Channels	A
2	Reactor Low Water Level	≥ +12" Indicated Level	X	X	X	4 Instrument Channels	A
2	High Water Level in Scram Discharge Volume	≤ 60 Gallons	X(2)	X	X	4 Instrument Channels	A
2	Main Steam Line High Radiation	≤ 3 X Normal Rated Power Background *(14)	X	X	X	4 Instrument Channels	A
4	Main Steam Line Isolation Valve Closure	≤ 10% Valve Closure	X(3)(13)	X(3)(13)	X (13)	8 Instrument Channels	A or C
2	Turbine Control Valve Fast Closure (Loss of Control oil Pressure)	Setpoint ≥ 800 psig Control Oil Pressure			X(4)	4 Instrument Channels	A or D
4	Turbine Stop Valve Closure	≤ 10% Valve Closure			X(4)	8 Instrument Channels	A or D
2	Turbine Control Valve - Fast Closure	Valve 10% closed			X(4)	4 Instrument Channels	A or D

*Alarm setting ≤ 1.5 X Normal Rated Power Background

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
	Item / Frequency b) Flow Rate Test-Each RHR service water pump shall deliver at least 2400 gpm at a TDH of 674 ft. or more. — after major pump maintenance and every 3 months
2. From and after the date that one of the RHR Service Water subsystem pumps is made or found to be inoperable for any reason, reactor operation must be limited to thirty days unless operability of that pump is restored within this period. During such thirty days all other active components of the RHR Service Water subsystem are operable.	2. When it is determined that one RHR Service Water pump is inoperable, the remaining components of that subsystem and the other subsystems shall be demonstrated to be operable immediately and daily thereafter.
3. From and after the date that one RHR Service Water Subsystem is made or found to be inoperable for any reason, reactor operation is limited to seven days unless operability of that subsystem is restored within this period. During such seven days all active components of the other RHR Service Water subsystem and its associated diesel-generators required for operation of such components (if no external source of power were available), shall be operable.	3. When on RHR Service Water subsystem becomes inoperable, the operable subsystem and the diesel-generators required for operation of such components shall be demonstrated to be operable immediately and daily thereafter.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT
D. HPCI Subsystem	D. HPCI Subsystem
1. The HPCI Subsystem shall be operable whenever there is irradiated fuel in the reactor vessel, reactor pressure is greater than 150 psig, and prior to reactor startup from a Cold Condition, except as specified in 3.5.D.2 and 3.5.D.3 below.	1. HPCI Subsystem testing shall be performed as follows:

	Item	Frequency
a.	Simulated Automatic Actuation Test	Once/operating cycle
b.	Pump Operability	Once/month
c.	Motor Operated valve Operability	Once/month
d.	Flow Rate at 1020 psig Reactor Vessel Pressure	Once/3 months
e.	Flow Rate test (Recirculate to condensate storage tank) at 150 psig Reactor Vessel Pressure	Once/operating cycle

The HPCI pump shall deliver at least 3000 gpm for a system head corresponding to a reactor pressure of 1020 to 150 psig.

7. Not required to be operable when primary containment integrity is not required.

8. Not required to be operable when the reactor pressure vessel head is not bolted to the vessel.

9. The APRM downscale trip is automatically bypassed when the IRM instrumentation is operable and not high.

10. To be considered operable, APRM's A, B, C and D must have at least 9 LPRM inputs while APRM's E and F must have at least 13 LPRM inputs. Additionally each APRM must have at least 2 LPRM inputs per level.

11. W is the recirculation loop flow in percent of rated.

12. See Subsection 2.1.A.1.

13. The design permits closure of any two lines without a scram being initiated.

14. The trip setting and alarm setting for the Main Steam Line High Radiation Monitor shall be ≤ 6 X and ≤ 3 X, respectively, Normal Rated Power Background during the period prior to achieving 50 per cent rated power for the first time. 1

TABLE 4.1-1

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT FUNCTIONAL TESTS
MINIMUM FUNCTIONAL TEST FREQUENCIES FOR SAFETY INSTRUMENT AND CONTROL CIRCUITS

	Group (2)	Functional Test	Minimum Frequency (3)
Mode Switch in Shutdown	A	Place Mode Switch in Shut down	Each refueling outage.
Manual Scram	A	Trip Channel and Alarm	Every 3 months
RPS Channel Test Switch	A	Trip Channel and Alarm	Every refueling outage or after channel maintenance
IRM			
High Flux	C	Trip Channel and Alarm (4)	One per week during refueling or startup and before each startup
Inoperative	C	Trip Channel and Alarm (4)	Once per week during refueling or startup and before each startup
APRM			
High Flux in Run	B	Trip Output Relays (4)	Once/week (While in Run Mode)
Inoperative	B	Trip Output Relays (4)	Once/week
Downscale *	B	Trip Output Relays (4)	Once/month (1)
Flow Bias	B	Trip Output Relays (4)	Once/month (1)
High Flux in Startup or Refuel	C	Trip Output Relays (4)	Once per week during refueling or startup and before each startup
High Reactor Pressure	A	Trip Channel Alarm	Every 1 month (1)

*With companion IRM Hi-Hi or Inoperable.

5. The water level in the reactor vessel will be perturbed and the corresponding level indicator changes will be monitored. This perturbation test will be performed every month after completion of the functional test program.

TABLE 4.1-2

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT CALIBRATION
MINIMUM CALIBRATION FREQUENCIES FOR REACTOR PROTECTION INSTRUMENT CHANNELS

Instrument Channel	Group (1)	Calibration (4)	Minimum Frequency
IRM High Flux	C	Comparison to APRM on Controlled Shutdowns	On Controlled Shutdown
APRM High Flux			
Output Signal	B	Heat Balance	Daily
Flow Bias Signal	B	With Standard Pressure Source	Every refueling
LPRM Signal	B	TIP System Traverse	Every 1,000 EFPH
High Reactor Pressure	A	Standard Pressure Source	Every 3 months
High Drywell Pressure	A	Standard Pressure Source	Every 3 months
Reactor Low Water Level	A	Pressure Standard	Every 3 months
High Water Level in Scram Discharge Volume	A	Water Column	Every refueling
Main Steam Line Isolation Valve Closure	A	Note (5)	Note (5)
Main Steam Line High Radiation	B	Standard Current Source (3)	Every 3 months
Turbine First Stage Pressure Permissive	A	Standard Pressure Source	Every 6 months
Turbine Control Valve Oil Pressure Trip	A	Standard Pressure Source	Once per operating cycle

1

Three APRM instrument channels are provided for each protection trip system. APRM's A and E operate contacts in one subchannel and APRM's C and E operate contacts in the other subchannel. APRM's B, D and F are arranged similarly in the other protection trip system. Each protection trip system has one more APRM than is necessary to meet the minimum number required per channel. This allows the bypassing of one APRM per protection trip system for maintenance, testing or calibration. Additional IRM channels have also been provided to allow for bypassing of one such channel. The bases for the scram setting for the IRM, APRM, high reactor pressure, reactor low water level, MSIV closure, generator load rejection and turbine stop valve closure are discussed in Specifications 2.1 and 2.2.

Instrumentation (pressure switches) for the drywell are provided to detect a loss-of-coolant accident and initiate the core standby cooling equipment. A high drywell pressure scram is provided at the same setting as the core standby cooling systems (CSCS) initiation to minimize the energy which must be accommodated during a loss-of-coolant accident and to prevent return to criticality. This instrumentation is a backup to the reactor vessel water level instrumentation.

High radiation levels in the main steam line tunnel above that due to the normal nitrogen and oxygen radioactivity is an indication of leaking fuel. A scram is initiated whenever such radiation level exceeds three times normal background. The purpose of this scram is to reduce the source of such radiation to the extent necessary to limit the amount of radioactivity released due to gross fuel failure. Discharge of excessive amounts of radioactivity to the environs is prevented by the air ejector offgas monitors which cause an isolation of the main condenser offgas line to the main stack.

The MSIV closure scram is set to scram when the isolation valves are 10% closed in 3 out of 4 lines. This scram anticipates the pressure and flux transient which would occur when the valves close. By scramming at this setting, the resultant transient is less severe than either the pressure or flux transients which would otherwise result.

A reactor mode switch is provided which actuates or bypasses the various scram functions appropriate to the particular plant operating status.

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

TABLE 3.2-A

INSTRUMENTATION THAT INITIATES PRIMARY CONTAINMENT ISOLATION

Minimum No. of Operable Instrument Channels Per Trip System (1)	Instrument	Trip Level Setting	Number of Instrument Channels Provided by Design	Action(2)
2 (6)	Reactor Low Water Level	$\geq$ +12" Indicated Level (3)	4 Inst. Channels	A
	Reactor Low Pressure (Shutdown Cooling Isolation)	$\leq$ 135 psig	2 Inst. Channels	C
2	Reactor Low-Low-Water Level	At or above -38.5" indicated level (4)	4 Inst. Channels	A
2 (6)	High Drywell Pressure	$\leq$ 2.0 psig	4 Inst. Channels	A
2	High Radiation Main Steam Line Tunnel	$\leq$ 3 X Normal Rated Power Background (8)	4 Inst. Channels	B
2	Low Pressure Main Steam Line	$\geq$ 880 psig (7)	4 Inst. Channels	B
2 (5)	High Flow Main Steam Line	$\leq$ 140% of Rated Steam Flow	4 Inst. Channels	B
2	Main Steam Line Tunnel/Turbine Bldg. High Temperature	$\leq$ 200 deg. F	4 Inst. Channels	B
1	Reactor Cleanup System High Diff. Flow	$\leq$ 40 gpmd	2 Inst. Channel	D
1	Reactor Cleanup System High-High Temperature	$\leq$ 140^{o}F	1 Inst. Channel	D

1

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	Action (2)
1	Reactor Cleanup Area Ambient High Temp.	$130^{o}F$	3 Inst. Channels	D
1	Reactor Cleanup Area Differential High Temp.	$\Delta 14^{o}F$	3 Inst. Channels	D

5. Two required for each steam line.

6. These signals also start SBGTS and initiate secondary containment isolation.

7. Only required in Run Mode (interlocked with Mode Switch).

8. The trip setting for the High Radiation Main Steam Line Tunnel shall be $\leq$ 6 X Normal Rated Power Background before the power level reaches 50 per cent rated power for the first time. 1

TABLE 3.2-B

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	Reactor Low-Low Water Level	≥ -38.5 in. indicated level	4 HPCI & RCIC Inst. Channels	Initiates HPCI & RCIC
2	Reactor Low-Low-Low Water Level	≥ -139.5 in. indicated level (4)	4 Core Spray & RHR Instrument Channels 4 ADS Instrument Channels	1. In conjunction with Low Reactor Pressure initiates Core Spray and LPCI 2. In conjunction with confirmatory low level High Drywell Pressure, 120 second time delay and LPCI or Core Spray pump interlock initiates Auto Blowdown (ADS) 3. Initiates starting of Diesel Generator
2	Reactor High Water Level	≤ +53 in. indicated level	2 Inst. Channels	Trips HPCI and RCIC turbines

TABLE 3.2-B (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)	2 Inst. Channels	Prevents inadvertent operation of containment spray during accident condition
2	Containment High Pressure	$1 < p < 2$ psig	4 Inst. Channels	Prevents inadvertent operation of containment spray during accident condition
1	Confirmatory Low Level	$\leq$ +12 in. indicated level	2 Inst. Channels	ADS Permissive
2	High Drywell Pressure	$\leq$ 2.0 psig	4 HPCI Inst. Channels	1. Initiates Core Spray LPCI; HPCI
			4 RHR & Core Spray Inst. Channels	2. Initiates starting of Diesel-Generator
2	Reactor Low Pressure	$\geq$ 450 psig	4 Inst. Channels	Permissive for open Core Spray and LPCI Injection valves. Coincident with high drywell pressure, start LPCI and Core Spray pumps

TABLE 3.2-B (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 Pressure	P $\leqslant$ 135 psig	2 Inst. Channels	In conjunction with PCIS signal permits closure of RHR (LPCI) injection valves
2	Reactor Low Pressure	$\geqslant$ 900 psig	4 Inst. Channels	Prevents actuation of LPCI break detection circuit (1 Recirc Pump Running)
2	High Drywell Pressure	$\leqslant$ 2.0 psig	4 Inst. Channels	1. In conjunction with Low-Low Reactor level, 120 second time delay and LPCI or Core Spray running, initiates Auto Blowdown (ADS)
1	Core Spray Pump Start Timer	5 sec	2 timers	In conjunction with loss of power initiates the starting of CSCS pumps.
1	LPCI pump Start Timer	10 sec 15 sec	2 timers 2 timers	In conjunction with loss of power initiates the starting of LPCI pumps.

TABLE 3.2-B (Continued)

INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

Minimum No. of Operable Instrument Channel Per Trip System (1)	Trip Function	Trip Level Setting	Number of Instrument Channels Provided by Design	Remarks
1	Auto Blowdown Timer	120 sec ± 5 sec	2 timers	In conjunction with Low Reactor Water Level, High Drywell Pressure and LPCI or Core Spray Pump running interlock, initiates Auto Blowdown
2	RHR (LPCI) Pump Discharge Pressure Interlock	≥100 psig	4 channels	Defers ADS actuation pending confirmation of Low Pressure core cooling system operation (LPCI or Core Spray Pump running interlock)
2	Core Spray Pump Discharge Pressure Interlock	100 ± 5 psig	4 channels	"
1	RHR (LPCI) Trip System bus power monitor	80% ± 10% of rated voltage (6)	2 Inst. Channels	Relay which continuously monitors availability of power to logic systems and annunciates upon loss of power
1	Core Spray Trip System bus power monitor	80% ± 10% of rated voltage (6)		"

TABLE 3.2-B (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	ADS Trip System bus power monitor	80% $\pm$ 10% of rated voltage (6)	3 Inst. Channels	Relay which continuously monitors availability of power to logic systems and annunciates upon loss of power
1	HPCI Trip System bus power monitor	"	2 Inst. Channels	
1	RCIC Trip System bus power monitor	"	2 Inst. Channels	"
2	Recirculation Pump A d/p	$\leq$ 2 psid	4 Inst. Channels	Operates RHR (LPCI) break detection logic which directs cooling water into unbroken recirculation loop
2	Recirculation Pump B d/p	$\leq$ 2 psid	4 Inst. Channels	
2	Recirculation Riser d/p A > B	$0.5 < p < 1.5$ psid	4 Inst. Channels	
1	Core Spray Sparger to Reactor Pressure Vessel d/p	0.74 psid	2 Inst. Channels	Alarm to detect core spray sparger pipe break
2	Condensate Storage Tank Low Level	$\geq$ 12" above tank bottom (10,000 gallons)	2 Inst. Channels	Provides interlock to HPCI pump suction valves

TABLE 3.2-B (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 High Level	$\leq$ 2" above normal water level	2 Inst. Channels	Transfers HPCI pump suction to suppression chamber
1	RCIC Turbine High Flow	$\pm$ 180" H_2O (2)	2 Inst. Channels	
2	RCIC Turbine Equipment Room High Ambient Temperature	$\leq$ 175 deg. F (2)	4 Inst.	
2	RCIC Vent High Differential Temperature	$\leq \Delta$ 50 deg. F (2)	4 Inst.	
2	RCIC Steam Line Low Pressure	100 $>$ P $>$ 50 psig (2)	4 Inst.	
1	HPCI Turbine Steam Line High Flow	$\pm$ 225" H_2O (3)	2 Inst. Channels	
2	Suppression Pool Area High Ambient Temperature	150°F	4 Inst. Channels	
2	Suppression Pool Area High Diff. Temperature	50°F	4 Inst. Channels	
1	HPCI Leak Detection Time Delay	15 min.	2 Inst.	

1

TABLE 3.2-B (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	30 min.	2 Inst.	
2 (5)	HPCI Steam Line Low Pressure	$100 > P > 50$ psig (3)	4 Inst.	
2	HPCI Equipment Room High Ambient Temperature	≤ 175 deg. F	4 Inst.	
2	HPCI Equipment Room High Diff. Temperature	$\leq \Delta 50$ deg. F (3)	4 Inst.	
1 per 4 kV Bus	4 kV Emergency Bus Undervoltage Relay	20% of Rated Voltage		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	65% of Rated Voltage		Permits sequential of vital loads
2 per 4kV Bus	Emergency Transformer Undervoltage	65% of Rated Voltage		1. Trips emergency transformer feed to 4KV emergency bus 2. Fast transfer permissive

TABLE 3.2-G

INSTRUMENTATION THAT INITIATES RECIRCULATION PUMP TRIP

Minimum Number of Operable Instrument Channels per Trip System (1)	Instrument	Trip Level Setting	Number of Instrument Channels Provided By Design	Action
1	Reactor High Pressure	$\leq$ 1135 psig	4	(2)
1	Reactor Low Water Level	$\geq$ -38.5 in. indicated level	4	(2)

NOTES FOR TABLE 3.2-G

1. Whenever the reactor is in the RUN Mode, there shall be one operable trip system for each parameter for each operating recirculation pump. If this cannot be met, the indicated action shall be taken.

2. Reduce power and place the mode selector-switch in a mode other than the RUN Mode.

TABLE 4.2-A

MINIMUM TEST AND CALIBRATION FREQUENCY FOR PCIS

Instrument Channel (5)	Instrument Functional Test	Calibration Frequency	Instrument Check
1) Reactor Low Pressure (Shutdown Cooling Permissive)	(1)	Once/3 months	None
2) Reactor Low-Low Water Level	(1)	Once/3 months	Once/day
3) Main Steam High Temp.	(1)	Once/operating cycle	Once/day
4) Main Steam High Flow	(1)	Once/3 months	None
5) Main Steam Low Pressure	(1)	Once/3 months	None
6) Reactor Water Cleanup High Flow (7)	(1)	Once/3 months	Once/day
7) Reactor Water Cleanup High Temp.(7)	(1)	Once/3 months	None
8) Reactor Cleanup Area High Temp. (8)	(1)	Once/Operating cycle	None
Logic System Functional Test (4) (6)			
1) Main Steam Line Isolation Valves Main Steam Line Drain Valves Reactor Water Sample Valves		Once/6 months	
2) RHR - Isolation Valve Control Shutdown Cooling Valves Head Spray		Once/6 months	
3) Reactor Water Cleanup Isolation		Once/6 months	

1

TABLE 4.2-F

MINIMUM TEST AND CALIBRATION FREQUENCY FOR SURVEILLANCE INSTRUMENTATION

Instrument Channel	Calibration Frequency	Instrument Check
1) Reactor Level	Once/6 months	Once Each Shift
2) Reactor Pressure	Once/6 months	Once Each Shift
3) Drywell Pressure	Once/6 months	Once Each Shift
4) Drywell Temperature	Once/6 months	Once Each Shift
5) Suppression Chamber Temperature	Once/6 months	Once Each Shift
6) Suppression Chamber Water Level	Once/6 months	Once Each Shift
7) Control Rod Position	NA	Once Each Shift
8) Neutron Monitoring (when in Startup or Run Mode)	Once Per Day (APRM Gain Adjust)	Once Each Shift

NOTES FOR TABLES 4.2-A THROUGH 4.2-F

1. Initially once every month. The compilation of instrument failure rate data may include data obtained from other boiling water reactors for which the same design instrument operates in an environment similar to that of DAEC. The failure rate data must be reviewed and approved by the AEC prior to any change in the once-a-month frequency.

2. Functional tests, calibrations and instrument checks are not required when these instruments are not required to be operable or are tripped. Functional tests shall be performed before each startup with a required frequency not to exceed once per week. Calibrations shall be performed prior to each startup or prior to controlled shutdowns with a required frequency not to exceed once per week. Instrument checks shall be performed at least once per day during those periods when the instruments are required to be operable.

3. This instrumentation is excepted from the functional test definition. The functional test will consist of injecting a simulated electrical signal into the measurement channel.

These instrument channels will be calibrated using simulated electrical signals.

4. Simulated automatic actuation shall be performed once each operating cycle. Where possible, all logic system functional tests will be performed using the test jacks.

5. Reactor low water level, high drywell pressure and high radiation main steam line tunnel are not included on Table 4.2-A since they are tested on Table 4.1-2.

6. The logic system functional tests shall include a calibration of time delay relays and timers necessary for proper functioning of the trip systems.

7. These signals are not PCIS trip signals but isolate the Reactor Water Cleanup system only.

8. This instrumentation is excepted from the functional test definition. The functional test will consist of comparing the analog signal of the active thermocouple element feeding the isolation logic to a redundant thermocouple element.

TABLE 4.2-G

MINIMUM TEST AND CALIBRATION FREQUENCY FOR RECIRCULATION PUMP TRIP

Instrument Channel	Instrument Functional Check	Calibration Frequency
Reactor High Pressure	Once/refueling cycle	Once/refueling cycle
Reactor Low Water Level	Once/refueling cycle	Once/refueling cycle

Logic System Functional Test	Frequency
Recirculation Pump Trip	Once/refueling cycle

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 6 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 14.6.2 of the FSAR.

Pressure instrumentation is provided to close the main steam isolation valves in RUN Mode when the main steam line pressure drops below 880 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 14.5 of the FSAR, therefore, closure of the Main Steam Isolation valves for thermal transient protection when not in RUN Mode is not required.

The HPCI high flow and temperature instrumentation are provided to detect a break in the HPCI steam piping. Tripping of this instrumentation results in actuation of HPCI isolation valves. Tripping logic for the high flow is a 1 out of 2 logic.

Temperature is monitored at two (2) locations with four (4) temperature sensors at each location. Two (2) sensors at each location are powered by "A" direct current control bus and two (2) by "B" direct current control bus. Each pair of sensors, e.g., "A" or "B", at each location are physically separated and the tripping of either "A" or "B" bus sensor will actuate HPCI isolation valves.

The trip settings of $\pm$ 225 inch of water which corresponds to 300% of design flow for high flow and 175°F and $\Delta 45^{\circ}$ for high temperature are such that core uncovery is prevented and fission product release is within limits.

The RCIC high flow and temperature instrumentation are arranged the same as that for the HPCI. The trip setting of $\pm$ 180" H_20 for high flow and 175° and $\Delta 45^{\circ}$ for temperature are based on the same criteria as the HPCI.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT

3.5 CORE AND CONTAINMENT COOLING SYSTEMS

Applicability:

Applies to the operational status of the core and suppression pool cooling subsystems.

Objective:

To assure the operability of the core and suppression pool cooling subsystems under all conditions for which this cooling capability is an essential re sponse

Specification:

A. Core Spray and LPCI Subsystems

1. Both core spray subsystems shall be operable whenever irradiated fuel is in the vessel and prior to reactor startup from a Cold Condition, except as specified in 3.5.A.2 and 3.5.G.3 below.

4.5 CORE AND CONTAINMENT COOLING SYSTEMS

Applicability:

Applies to the Surveillance Requirements of the core and suppression pool cooling subsystems which are required when the corresponding Limiting Condition for operation is in effect.

Objective:

To verify the operability of the core and suppression pool cooling subsystems under all conditions for which this cooling capability is an essential response to station abnormalities.

Specification:

A. Core Spray and LPCI Subsystems

1. Core Spray Subsystem Testing.

	Item	Frequency
a.	Simulated Automatic Actuation test.	Once/Operating Cycle
b.	Pump Operability	Once/month
c.	Motor-Operated Valve Operability	Once/month

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT
	Item — **Frequency** d. Pump flow rate - Both loops shall deliver at least 3020 gpm against a system head corresponding to a reactor vessel pressure of 113 psig. — Once/3 months
2. From and after the date that one of the core spray subsystems is made or found to be inoperable for any reason, continued reactor operation is permissible during the succeeding seven days provided that during such seven days all active components of the other core spray subsystem and active components of the LPCI subsystem and the diesel generators are operable.	2. When it is determined that one core spray subsystem is inoperable, the operable core spray subsystem, the LPCI subsystem and the diesel generators shall be demonstrated to be operable immediately. The operable core spray subsystem shall be demonstrated to be operable daily thereafter.
3. The LPCI Subsystems shall be operable whenever irradiated fuel is in the reactor vessel, and prior to reactor startup from a Cold Condition, except as specified in 3.5.A.4, 3.5.A.5 and 3.5.G.3 below.	3. LPCI Subsystem Testing shall be as follows: a. Simulated Automatic Actuation Test — Once/Operating Cycle b. Pump Operability — Once/month

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
	c. Whenever work is performed that could adversely affect the filter system efficiency and at least once each operating cycle it shall be demonstrated that:
	1) The removal efficiency of the particulate filters is not less than 99% for particulate matter larger than 0.3 micron.
	2) The removal efficiency of each of the charcoal filters is not less than 99% for iodine.
	d. At least once each five years representative in-place samples shall be removed and adsorption shall be demonstrated.
	e. At least once per operating cycle automatic initiation of each branch of the standby gas treatment system shall be demonstrated.
	f. At least once per operating cycle manual operability of the bypass system for filter cooling shall be demonstrated.
2. From and after the date that one train of the standby gas treatment system is made or found to be inoperable for any reason, continued reactor operation or fuel handling is permissible only during the succeeding seven days unless such train is sooner made operable, provided that during such seven days all active components of the other standby gas treatment train shall be operable.	2. When one train of the standby gas treatment system becomes inoperable, the operable train shall be demonstrated to be operable immediately and daily thereafter.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3. If Specifications 3.7.B.1 and 3.7.B.2 are not met, the reactor shall be placed in the cold shutdown condition and fuel handling operations shall be prohibited.	

through the filter media. Considering the relatively short time that the fans may be run for test purposes, plugging is unlikely, and the test interval of once per operating cycle is reasonable. Duct heater tests will be conducted once during each operating cycle. Considering the simplicity of the heating circuit, the test frequency is sufficient.

The in-place testing of charcoal filters is performed using Freon-11 or equivalent, which is injected into the system upstream of the charcoal filters. Measurements of the Freon concentration upstream and downstream of the charcoal filters are made. The ratio of the inlet and outlet concentrations gives an overall indication of the leaktightness of the system. Although this is basically a leak test, since the filters have charcoal of known efficiency and holding capacity for elemental iodine and/or methyl iodine, the test also gives an indication of the relative efficiency of the installed system.

High efficiency particulate filters are installed before and after the charcoal filters to minimize potential release of particulates to the environment and to prevent clogging of the iodine filters. An efficiency of 99% is adequate to retain particulates that may be released to the reactor building following an accident. This will be demonstrated by in-place testing with DOP as testing medium. An 11 kw heater maintains relative humidity below 70% in order to assure the efficient removal of methyl iodine on the impregnated charcoal filters.

The test interval for filter efficiency was selected to minimize plugging of the filters. In addition, retention capacity in terms of milligrams of iodine per gram of charcoal will be demonstrated. This will be done by removing grab samples from the charcoal bed. These tests will normally be performed every five years unless filter efficiency seriously deteriorates. Since shelf lives greater than five years have been demonstrated, the test interval is reasonable.

8. Primary Containment Power Operated Isolation Valves

Automatic isolation valves are provided on process piping which penetrates the containment and communicates with the

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.8 AUXILIARY ELECTRICAL SYSTEM	4.8 AUXILIARY ELECTRICAL SYSTEM
Applicability:	Applicability:
Applies to the auxiliary electrical power system.	Applies to the periodic testing requirements of the auxiliary electrical systems.
Objective:	Objective:
To assure an adequate supply of electrical power for operation of those systems required for safety.	Verify the operability of the auxiliary electrical system.
Specification:	Specification:
A. Auxiliary Electrical Equipment	A. Auxiliary Electrical Equipment
The reactor shall not be made critical unless all of the following conditions are satisfied:	
1. Both off-site sources and the startup transformers and standby transformers are available and capable of automatically supplying power to the 4kV emergency buses.	1. Diesel-Generators a. Each diesel-generator shall be manually started and loaded once each month to demonstrate operational readiness. The test shall continue for at least a one-hour period at rated load.
2. The two diesel-generators shall be operable and there shall be a minimum of 35,000 gallons of diesel fuel in the diesel fuel oil tank.	During the monthly generator test the diesel-generator starting air compressor shall be checked for operation and its ability to recharge air receivers. The operation of the diesel fuel oil transfer pumps shall be demonstrated, and the diesel starting time to reach rated voltage and frequency shall be logged.
3. All station 24,125 and 250 volt battery systems shall be operable. The associated battery chargers for the 24 volt batteries, two of the three battery	

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT
chargers for the 125 volt station batteries, and one of the two 250 volt battery chargers shall be operable.	b. Once per operating cycle the condition under which the diesel-generator is required will be simulated and a test conducted to demonstrate that it will start and accept the emergency load within the specified time sequence. The results shall be logged.
4. The emergency 4160 volt buses 1A3 and 1A4 and 450 volt buses 1B3, 1B4, 1B9 and 1B20 shall be energized and operable.	c. The quantity of diesel fuel available shall be logged monthly and after each use of the diesels.
	d. Once a month a sample of diesel fuel shall be checked for viscosity, water and sediment. The values for viscosity, water and sediment shall be within the acceptable limits specified in Table 1 of ASTM D975-68 and logged.
	e. Each diesel-generator shall be given an annual inspection in accordance with instructions based on the manufacturer's recommendations.
	f. A sample test and record shall be made of each oil delivery before it is placed in the storage tank.
	2. Unit Batteries
	a. Every week the specific gravity, the voltage and temperature of the pilot cell and overall battery voltage shall be measured and logged.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
	b. Every three months the measurements shall be made of voltage of each cell to nearest 0.01 Volt, specific gravity of each cell, and temperature of every fifth cell. These measurements shall be logged.
	c. Once each operating cycle, the stated batteries shall be subjected to a rated load discharge test. The specific gravity and voltage of each cell shall be determined after the discharge and logged.
B. <u>Operation with Inoperable Components</u>	B. <u>Surveillance with Inoperable Components</u>
Whenever the reactor is in Run Mode or Startup Mode with the reactor not in the Cold Condition, the requirements of 3.8.A shall be met except:	
1. Diesel-Generators	1. Diesel-Generators
From and after the date that one of the diesel-generators or its associated buses are made or found to be inoperable for any reason and the remaining diesel-generator is operable, the requirements of Specification 3.5.G.1 shall be satisfied.	When it is determined that one of the diesel-generators is inoperable the requirements of Specification 4.5.G.1 shall be satisfied.
2. Batteries	2. Batteries
a. From and after the date that ventilation is lost in the battery room, portable ventilation equipment shall be provided.	a. From and after the date that ventilation is lost in the battery room, samples of the battery room atmosphere shall be taken daily for hydrogen concentration determination.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
b. From and after the date that one of the two 125 volt station battery systems is made or found to be inoperable for any reason, reactor operation may continue for three days provided Specification 3.5.G is met, and repair is immediately initiated.	
c. From and after the date that the 250 volt station battery system is made or found to be inoperable for any reason, the HPCI system shall be considered to be inoperable and the requirements of Specification 3.5.D shall be met.	
3. <u>Offsite Power</u>	3. <u>Offsite Power</u>
a. From and after the date that the startup or standby transformer and one diesel-generator or associated buses are made or found to be inoperable for any reason, reactor operation may continue provided the requirements of Specification 3.5.G.1 are satisfied.	a. When it is determined that one of the diesel-generators or associated buses is inoperable, the requirements of Specification 4.5.G.1 shall be satisfied.
b. From and after the date that both the startup and standby transformers become inoperable, reactor operation may continue for seven days provided both emergency diesel-generators, associated buses and all low pressure cooling systems are operable.	b. When it is determined that both the startup and standby transformers are inoperable both diesel-generators, associated buses and all low pressure core and containment cooling systems shall be demonstrated to be operable immediately and daily thereafter.

4.8 BASES:

The monthly tests of the diesel-generators are conducted to demonstrate satisfactory system performance and operability. The test of the automatic starting circuits will prove that each diesel will receive all automatic start signals. The loading of each diesel-generator is conducted to demonstrate proper operation at maximum expected emergency loading and at equilibrium operating conditions. Generator experience at other generator stations indicates that the testing frequency is adequate to assure a high reliability of operation should the system be required.

Each diesel-generator has two independent starting air supply systems. One consists of a motor driven air compressor which automatically recharges two air receivers and the other consists of a diesel driven air compressor which is manually operated to recharge a third air receiver. During the monthly check of the diesel-generator, both air start systems will be checked for proper operation.

Following the tests (at least monthly) or other operation of the units, the fuel volume remaining in the diesel oil storage tank will be checked.

At the end of the monthly loads test of the diesel-generator, the fuel oil transfer pump will be operated to refill the day tank and to check the operation of this pump.

The day tank level indicator and alarm switches and fuel oil transfer pump control switches will be checked at this time.

The test of the diesels once each operating cycle will be more comprehensive in that it will functionally test the system; i.e., it will check starting and closure of breakers and sequencing of loads. The units will be started by simulation of a loss-of-coolant accident. In addition, a loss of normal power condition will be imposed to simulate a loss of offsite power. The timing sequence will be checked to assure proper loading in the time required. Periodic tests check the capability of the units to start in the required time and to deliver the expected emergency load requirements. Periodic testing of the various components plus a functional test each operating cycle are sufficient to maintain adequate reliability.

Logging the diesel fuel supply after each operation (at least monthly) assures that the minimum fuel supply requirements will be

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT
3.9 CORE ALTERATIONS	**4.9 CORE ALTERATIONS**
<u>Applicability:</u>	<u>Applicability:</u>
Applies to the fuel handling and core reactivity limitations.	Applies to the periodic testing of those interlocks and instrumentation used during refueling and core alterations.
<u>Objective:</u>	<u>Objective:</u>
To ensure that core reactivity is within the capability of the control rods and to prevent criticality during refueling.	To verify the operability of instrumentation and interlocks used in refueling and core alterations.
<u>Specification:</u>	<u>Specification:</u>
A. <u>Refueling Interlocks</u>	A. <u>Refueling Interlocks</u>
1. The reactor mode switch shall be locked in the "Refuel" position during core alterations and the refueling interlocks shall be operable except as specified in 3.9.A.5 and 3.9.A.6 below.	1. Prior to any fuel handling with the head off the reactor vessel, those refueling interlocks applicable to the equipment being used shall be functionally tested. They shall be tested at weekly intervals thereafter until no longer required. They shall also be tested following any repair work associated with the interlocks.
2. Fuel shall not be loaded into the reactor core unless all control rods are fully inserted except as modified by Specification 3.9.A.6.a.	2. Prior to performing control rod or control rod drive maintenance on control cells without removing fuel assemblies, the directional control valves shall be electrically disarmed at least on the other control rod drives in the 5 x 5 rod array centered on the control rod or rod drive undergoing maintenance. Then it shall be demonstrated that the core can be made subcritical by a margin of 0.38 percent

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT
	Δk at any time during the maintenance with the strongest remaining operable control rod fully withdrawn and all other operable rods fully inserted. Alternatively if the remaining control rods are fully inserted and have had their directional control valves electrically disarmed, it is sufficient to demonstrate that the core is subcritical with a margin of at least 0.38% Δk at any time during the maintenance. A control rod on which maintenance is being performed shall be considered inoperable.
3. The fuel grapple hoist load switch shall be set at ≤ 400 lbs.	
4. If the frame-mounted auxiliary hoist, the monorail-mounted auxiliary hoist, or the service platform hoist is to be used for handling fuel with the head off the reactor vessel, the load limit switch on the hoist to be used shall be set at ≤ 400 lbs.	
5. A maximum of two nonadjacent control rods may be withdrawn from the core for the purpose of performing control rod and/or control rod drive maintenance, provided the following conditions are satisfied:	
a. The reactor mode switch shall be locked in the "refuel" position. The refueling interlock which prevents more than one control rod from being withdrawn may be bypassed for one of the control rods on which maintenance is being performed. All other refueling interlocks shall be operable.	

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT
control rod after the fuel assemblies in the cell containing (controlled by) that control rod have been removed from the reactor core. All other refueling interlocks shall be operable.	

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT
b. A sufficient number of control rods shall be operable so that the core can be made subcritical with the strongest operable control rod fully withdrawn and all other operable control rods fully inserted, or all directional control valves for remaining control rods shall be disarmed electrically and sufficient margin to criticality shall be demonstrated.	
c. If maintenance is to be performed on two control rod drives, they must be separated by more than two control cells in any direction.	
d. An appropriate number of SRM's are available as defined in specification 3.9.B.	
e. The directional control valves of the control rod drives in the 5 x 5 array centered on the control rod or drive undergoing maintenance must be electrically disarmed.	
6. Any number of control rods may be withdrawn or removed from the reactor core providing the following conditions are satisfied:	
a. The reactor mode switch is locked in the "refuel" position. The refueling interlock which prevents more than one control rod from being withdrawn may be bypassed on a withdrawn	

if a control rod is withdrawn and fuel is on a hoist. Likewise, if the refueling platform is over the core with fuel on a hoist, control rod motion is blocked by the interlocks. When the mode switch is in the refuel position, only one control rod can be withdrawn. The refueling interlocks, in combination with core nuclear design and refueling procedures, limit the probability of an inadvertent criticality. The nuclear characteristics of the core assure that the reactor is subcritical even when the highest worth control rod is fully withdrawn. The combination of refueling interlocks for control rods and the refueling platform provide redundant methods of preventing inadvertent criticality even after procedural violations. The interlocks on hoists provide yet another method of avoiding inadvertent criticality.

Fuel handling is normally conducted with the fuel grapple hoist. The total load on this hoist when the interlock is required consists of the weight of the fuel grapple and the fuel assembly. This total is approximately 900 lbs., in comparison to the load-trip setting of 400 lbs. Provisions have also been made to allow fuel handling with either of the three auxiliary hoists and still maintain the refueling interlocks. The 400 lb. load-trip setting on these hoists is adequate to trip the interlock when one of the more than 600-lb. fuel bundles is being handled.

During certain periods, it is desirable to perform maintenance on two control rods and/or control rod drives at the same time. The maintenance is performed with the mode switch in the "refuel" position to provide the refueling interlocks normally available during refueling operations. In order to withdraw a second control rod after withdrawal of the first rod, it is necessary to bypass the refueling interlock on the first control rod which prevents more than one control rod from being withdrawn at the same time. The requirement that an adequate shutdown margin be demonstrated or that all remaining control rods have their directional control valves electrically disarmed ensures that inadvertent criticality cannot occur during this maintenance. The adequacy of the shutdown margin is verified by demonstrating that the core is shut down by a margin of 0.38 percent Δk with the strongest operable control rod fully withdrawn, or that at least 0.38 percent Δk shutdown margin is available if the remaining control rods have had their directional control valves disarmed. Disarming the directional control valves does not inhibit control rod scram capability.

Specification 3.9.A.6 allows unloading of a significant portion of the reactor core. This operation is performed with the mode switch in the "refuel" position to provide the refueling interlocks

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT
3.10 ADDITIONAL SAFETY RELATED PLANT CAPABILITIES	4.10 ADDITIONAL SAFETY RELATED PLANT CAPABILITIES
<u>Applicability</u>:	<u>Applicability</u>:
Applies to the operating status of the main control room ventilation system and the emergency shutdown local control panel.	Applies to the surveillance requirements for the main control room ventilation system, and the emergency shutdown control panels which are required by the coreesponding Limiting Conditions for Operation.
<u>Objective</u>:	<u>Objective</u>:
To assure the availability of the main control room ventilation system, and emergency shutdown control panels under the conditions for which the capability is an essential response to station abnormalities.	To verify that operability or availability under conditions for which these capabilities are an essential response to station abnormalities.
A. <u>Main Control Room Ventilation</u>	A. <u>Main Control Room Ventilation</u>
1. The reactor shall not have a coolant temperature greater than 212°F and fuel may not be handled unless bcth of the control room ventilation fans and standby filter units and control room radiation monitors are available for normal operation except that one ventilation fan, filter unit and/or radiation monitor may be out of service for 1 month.	1. Each of the control room ventilation air supply fans and dampers shall be tested for operability every 3 months. The standby filter units shall be tested once per operating cycle as follows: a. Pressure drop test across each filter and the filter system. b. Demonstrate that the removal efficiency of the particulate filters is not less than 99% for particulate matter larger than 0.3 micron. 1

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT
	c. Freon-11 test for charcoal filter bypass as a measure of filter efficiency of at least 99% for halogen removal.
	d. A sample of the charcoal filter shall be analyzed once per 5 years to assure halogen removal efficiency of at least 99%.
	e. Operability of the main control room air intake radiation monitors shall be tested every 3 months.
B. <u>Emergency Shutdown Control Panel</u>	B. <u>Emergency Shutdown Local Control Panel</u>
1. At all times when not in use or being maintained, the emergency shutdown local control panel shall be secured.	1. The emergency shutdown local control panel shall be visually checked once per week to verify it is secured.
	2. Operability of the switches on the emergency shutdown local control panel shall be functional tested once per refueling outage.

3.10 BASES:

1. Main Control Room Ventilation System

One main control room ventilation fan provides adequate ventilation flow under accident conditions. Should one ventilation fan and/or standby filter unit be out of service during reactor operation, the allowable repair time of 1 month is justified, based on the 3 month test interval.

At least one channel (detector) in the Control Room Ventilation Intake Radiation Monitoring System must be operable at all times for indication-alarm of radioactive air being drawn into the main control room. Each control room intake air filtration train starts when a trip signal from the detectors is given via failure or isolate signal from its respective channel. 1

2. Emergency Shutdown Local Control Panel

The Emergency Shutdown Local Control Panel is provided to assure the capability of controlling reactor pressure for taking the plant to the hot shutdown condition external to the control room for the unlikely condition that the control room becomes uninhabitable.

4.10 BASES:

1. Main Control Room Ventilation System

The 3-month test interval for the main control room ventilation fan and dampers is based on good engineering judgement since two redundant trains are provided and both are normally in operation.

A pressure drop test across each filter and across the filter system is a measure of filter system condition. DOP injection measures particulate removal efficiency of the high-efficiency particulate filters. A Freon-11 test of the charcoal filters is essentially a leakage test. Since the filters have charcoal of known efficiency and holding capacity for elemental iodine and/or methyl iodine, the test also gives an indication of the relative efficiency of the installed system. Laboratory analysis of a sample of the charcoal filters positively demonstrates halogen removal efficiency. These tests are conducted in accordance with manufacturers' recommendations.

2. Emergency Shutdown Local Control Panel

Once per week verification of the panel being properly secured is considered adequate. The associated equipment is proven operable during surveillance testing of that equipment. An operability verification by functional test at each refueling outage is adequate to assure that the panel is available and can perform its design function.

6.0 ADMINISTRATIVE CONTROLS

6.1 MANAGEMENT - AUTHORITY AND RESPONSIBILITY

6.1.1 The Chief Engineer has primary responsibility for the safe operation of the DAEC-1 plant, and reports, under the Executive Vice President to the General Production Manager.

6.5 REVIEW AND AUDIT

6.5.1 Operations Committee

A committee of technically qualified plant staff members shall be appointed by the Chief Engineer to perform timely and continuing reviews of plant operations. The committee's activities shall be governed by a written charter which shall include:

1. Specification of committee membership and designation of its chairman. The qualifications of the regular members shall be maintained at a level equal to or greater than that specified in ANSI N18.1-1971.

2. The administrative procedures by which it functions, including specification of a quorum, meeting frequency, maintenance of records, and transmitting of its decisions.

3. Its authority.

4. Its review responsibilities.

    a. Review of all abnormal occurrences.

b. Review of all proposed normal, abnormal and emergency operating procedures and changes to these procedures. Review of proposed procedures and changes relative to Maintenance, Repair and Replacement, Radiation Protection, Reactor Core Component Handling, Surveillance Testing, Administrative Control, Security, and Preparedness. Review of any other proposed procedures or changes as determined by the Chief Engineer to affect nuclear safety.

c. Review proposed tests and experiments when such activities may constitute an unreviewed safety question.

d. Review proposed changes to Technical Specifications, Operating License and the FSAR.

e. Review proposed changes or modifications to plant systems or equipment having safety-related functions which changes would require a change

2. Membership

Membership in the Safety Committee shall be by appointment by the Executive Vice President, and shall consist of eight (8) persons, two of whom shall be designated as Chairman and Vice-Chairman, respectively. Not more than a minority of a quorum may have concurrent on-site line responsibility for the operation of the DAEC and no such member shall be elegible to be Chairman or Vice Chairman.

3. Qualifications of Membership

Members of the Safety Committee shall collectively have or have access to applicable technical and experience expertise in the following areas:

a. Nuclear power plant operations
b. Nuclear engineering
c. Chemistry and Radiochemistry
d. Instrumentation and Control
e. Radiation Protection
f. Mechanical and Electrical Engineering
g. Nuclear Safety

4. Quorum

A quorum shall consist of not less than a majority of the voting members of the Safety Committee.

5. Initiating Safety Committee Activities

The charter shall specify the mechanisms whereby subjects requiring review and subjects selected for audit by the Safety Committee are brought to its attention.

6. Meeting Frequency

The Safety Committee shall meet on call by the Chairman or at the request of any member and at least semi-annually following the first year of plant operation. Meeting frequency during the first year of plant operation shall not be less than once per calendar quarter.