PROPOSED CHANGE RTS 121 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 licnese by deleting current pages and replacing them with the attached new proposed pages. A list of affected pages is included.

1. The more detailed definition of the term OPERABLE as requested by the NRC in a letter dated April 10, 1980 has been incorporated. Incorporation of the general LCO, as suggested by the letter, was not included to avoid confusion that might result from differences between the general LCO and LCO's found in the individual specifications.

2. Provisions for the single recirculation loop operation allowed by License Amendment No. 60 has been deleted.

3. The plant staffing section of the Technical Specification has been updated to reflect the new organizational changes which have been made to further enhance managerial control of safe operation of the DAEC.

4. Section 1.0 was retyped to eliminate blank spaces and pages created by previous changes of material.

5. An administrative error was corrected on page 1.1-3. The scram and isolation for reactor low water level was changed from 513.5 to 514.5 inches to correct an error that was introduced in the RTS-117 submittal and included in Amendment No. 59 to the License.

6. The primary containment isolation instrumentation table (3.2-A) has been updated to include a clarification note. This clarification was requested by on-site NRC inspectors.

7. The change "Table" to "Figures" on page vii corrects an administrative error.

8. The remainder of the changes, except for one page which was retyped to insure clarity, consist exclusively of renumbering pages to eliminate blank pages created by previous changes of material.

List of Pages To Be Deleted

vii **	1.1-26	3.6-12 *
1.0-1 **	1.1-27	3.6-13 *
1.0-2 **	1.1-28	3.6-14 *
1.0-3 *	1.2-7	6.1-1 **
1.0-4 *	3.2-5.a**	6.2-4 **
1.0-5 *	3.2-16 **	6.4-1 **
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1.0-9	3.5-29	6.9-2 *
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1.0-11	3.5-31	6.9-4 *
1.0-11a	3.5-32	6.9-5
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* To be replaced by an administratively changed/renumbered page

** Replacement page contains actual change to Technical Specifications or correction of previous administrative error.

TECHNICAL SPECIFICATIONS

LIST OF FIGURES

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 Atomic Energy 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), components(s) and device(s) are OPERABLE, or likewise satisfy the requirements of this specification.

4. DELETED

5. OPERABLE-OPERABILITY

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

6. OPERATING

Operating means that a system or component is performing its intended functions in its required manner.

7. IMMEDIATE

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

8. REACTOR POWER OPERATION

Reactor power operation is any operation with the mode switch in the "Startup" or "Run" position with the reactor critical and above 1% rated power.

9. HOT STANDBY CONDITION

Hot standby condition means operation with coolant temperature greater than 212°F, reactor vessel pressure less than 1035 psig, and the mode switch in the Startup/Hot Standby position.

10. COLD CONDITION

Reactor coolant temperature equal to or less than 212°F.

11. HOT SHUTDOWN

The reactor is in the shutdown mode and the reactor coolant temperature greater than 212°F.

12. COLD SHUTDOWN

The reactor is in the shutdown mode, the reactor coolant temperature equal to or less than 212°F, and the reactor vessel is vented to atmosphere.

13. MODE OF OPERATION

A reactor mode switch selects the proper interlocks for the operational status of the unit. The following are the modes and interlocks provided:

a. Startup/Hot Standby Mode - In this mode the reactor protection scram trips, initiated by main steam line isolation valve closure, are bypassed when reactor pressure is less than 1035 psig, the reactor protection system is energized with IRM neutron monitoring system trip, the APRM 15% high flux trip, and control rod withdrawal interlocks in service. The lower pressure MSIV closure 880 psig trip is also bypassed. This is intended to imply the Startup/Hot Standby position of the mode switch.

b. Run Mode - In this mode the reactor vessel pressure is at above 880 psig and the reactor protection system is energized with APRM protection (excluding the 15% high flux trip) and RBM interlocks in service.

c. Shutdown Mode - Placing the mode switch to the shutdown position initiates a reactor scram and power to the control rod drives is removed. After a short time period (about 10 sec), the scram signal is removed allowing a scram reset and restoring the normal valve lineup in the control rod drive hydraulic system; also, the main steam line isolation scram is bypassed if reactor vessel pressure is below 1035 psig.

d. Refuel Mode - With the mode switch in the refuel position interlocks are established so that one control rod only may be withdrawn when the Source Range Monitor indicates at least 3 cps and the refueling crane is not over the reactor; also, the main steam line isolation scram is bypassed if reactor vessel pressure is below 1035 psig. If the refueling crane is over the reactor, all rods must be fully inserted and none can be withdrawn.

14. DESIGN AND RATED POWER

Rated power (100% power) refers to operation at a reactor power of 1593 Mwt. Design power, the power to which the safety analysis applies is 105% of rated steam flow which corresponds to 1658 Mwt.

15. PRIMARY CONTAINMENT INTEGRITY

Primary Containment Integrity means that the drywell and pressure suppression chamber are intact and all of the following conditions are satisfied:

a. All nonautomatic containment isolation valves on lines connected to the reactor coolant system or containment which are not required to be open during accident conditions are closed. These valves may be opened to perform necessary operational activities.

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

c. All automatic containment isolation valves are operable or deactivated in the isolated position.

d. All blind flanges and manways are closed.

16. SECONDARY CONTAINMENT INTEGRITY

Secondary containment integrity means that the reactor building is intact and the following conditions are met:

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

b. The standby gas treatment system is operable.

c. All Reactor Building ventilation system automatic isolation valves are operable or deactivated in the isolated position.

17. OPERATING CYCLE

Interval between the end of one refueling outage and the end of the next subsequent refueling outage.

18. REFUELING OUTAGE

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

19. ALTERATION OF THE REACTOR CORE

The act of moving any component in the region above the core support plate, below the upper grid and within the shroud. Normal control rod movement with the control drive hydraulic system is not defined as a core alteration. Normal movement of in-core instrumentation and the transversing in-core probe is not defined as a core alteration.

20. REACTOR VESSEL PRESSURE

Unless otherwise indicated, reactor vessel pressures listed in the Technical Specifications are those measured by the reactor vessel steam space detectors.

21. THERMAL PARAMETERS

a. Minimum Cirtical Power Ratio (MCPR) - The value of critical power ratio (CPR) for that fuel bundle having the lowest CPR.

b. Critical Power Ratio (CPR) - The ratio of that fuel bundle power which would produce boiling transition to the actual fuel bundle power.

c. Transition Boiling - Transition boiling means the boiling regime between nucleate and film boiling. Transition boiling is the regime in which both nucleate and film boiling occur intermittently with neither type being completely stable.

d. Deleted

e. Linear Heat Generation Rate - the heat output per unit length of fuel pin.

f. Fraction of Limiting Power Density (FLPD) - The fraction of limiting power density is the ratio of the linear heat generation rate (LHGR) existing at a given location to the design LHGR for that bundle type.

g. Maximum Fraction of Limiting Power Density (MFLPD) - The maximum fraction of limiting power density is the highest value existing in the core of the fraction of limiting power density (FLPD).

h. Fraction of Rated Power (FRP) - The fraction of rated power is the ratio of core thermal power to rated thermal power of 1593 MWth.

22. INSTRUMENTATION

a. Instrument Calibration - An instrument calibration means the verification or adjustment of an instrument signal output so that it corresponds, with acceptable range, and accuracy, to a known value(s) of the parameter which the instrument monitors. The acceptable range and accuracy of an instrument and its setpoint are given in the system design control document and these setpoints are used in the Technical Specifications.

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

c. Instrument Functional Test - An instrument functional test means the injection of a simulated signal into the instrument primary sensor to verify the proper instrument channel response, alarm and/or initiating action.

d. Instrument Check - An instrument check is qualitative determination of acceptable operability by observation of instrument behavior during operation. This determination shall include, where possible, comparison of the instrument with other independent instruments measuring the same variable.

e. Logic System Functional Test - A logic system functional test means a test of all relays and contacts of a logic circuit to insure all components are operable per design intent. Where practicable, action will go to completion; i.e., pumps will be started and valves operated.

f. Trip System - A trip system means an arrangement of instrument channel trip signals and auxiliary equipment required to initiate action to accomplish a protective trip function. A trip system may require one or more instrument channel trip signals related to one or more plant parameters in order to initiate trip system action. Initiation or protective action may require the tripping of a single trip system or the coincident tripping of two trip systems.

g. Protection Action - An action initiated by the protection system when a limit is reached. A protective action can be at a channel or system level.

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 arrangment 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 FSAR Subsection 14.5).

23. FUNCTIONAL TESTS

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.

26. SURVEILLANCE FREQUENCY

Periodic surveillance tests, checks, calibrations and examinations shall be performed within the specified suveillance intervals. These intervals may be adjusted plus or minus 25%. The operating cycle interval as pertaining to instrument and electrical surveillance shall never exceed 15 months. In cases where the elapsed interval has exceeded 100% of the specified interval, the next surveillance interval shall commence at the end of the original specified interval.

27. FIRE SUPPRESSION WATER SYSTEM

A fire suppression water system shall consist of a water source, pumps, and distribution piping with associated sectionalizing control or isolation valves. Such valves include yard hydrant curb valves, the first valve ahead of the water flow alarm device on each sprinkler, hose standpipe or deluge system riser.

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING
C. Power Transients To ensure that the Safety Limits established in Specification 1.1.A and 1.1.B are not exceeded, each required scram shall be initiated by its primary source signal. A Safety Limit shall be assumed to be exceeded when scram is accomplished by a means other than the Primary Source Signal. D. With irradiated fuel in the reactor vessel, the water level shall not be less than 12 in. above the top of the normal active fuel zone. Top of the active fuel zone is defined to be 344.5 inches above vessel zero (see Bases 3.2).	Where: S = Setting in percent of rated power (1,593 MWt) W = Recirculation loop flow in percent of rated flow. Rated recirculation loop flow is that recirculation loop flow which corresponds to 49×10^6 lb/hr core flow. For a MFLPD greater than FRP, the APRM scram setpoint shall be: $S \leq (0.66\ W + 54)\ \frac{FRP}{MFLPD}$ NOTE: These settings assume operation within the basic thermal design criteria. These criteria are LHGR $\leq$ 18.5 KW/ft (7x7 array) or 13.4 KW/ft (8x8 array) and MCPR $\geq$ values as indicated in Table 3.12-2 times K_f, where K_f is defined by Figure 3.12-1. Therfore, at full power, operation is not allowed with MFLPD greater than unity even if the scram setting is reduced. If it is determined that either of these design criteria is being violated during operation, action must be taken immediately to return to operation within these criteria. 2. APRM High Flux Scram When in the REFUEL or STARTUP and HOT STANDBY MODE. The APRM scram shall be set at less than or equal to 15 percent of rated power.

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING
	3. APRM Rod Block when in Run Mode. For operation with MFLPD less than or equal to FRP the APRM Control Rod Block setpoint shall be as shown on Fig. 2.1-1 and shall be: $S \leq (0.66\ W + 42)$ The definitions used above for the APRM scram trip apply. For a MFLPD greater than FRP, the APRM Control Rod Block setpoint shall be: $S \leq (0.66\ W + 42)\ \frac{FRP}{MFLPD}$
	4. IRM - The IRM scram shall be set at less than or equal to 120/125 of full scale.
	B. Scram and Isolation on reactor low water level — $\geq$514.5 inches above vessel zero (+12" on level instruments)
	C. Scram - turbine stop valve closure — $\leq$ 10 percent valve closure
	D. Turbine control valve fast closure shall occur within 30 milliseconds of the start of turbine control valve fast closure.

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°F	3 Inst. Channels	D
1	Reactor Cleanup Area Differential High Temp.	Δ14°F*	3 Inst. Channels	D
2	Loss of Main Condensor Vacuum	≤10 in Hg Vacuum	4 Inst. Channels	B

* Note: The actual set point shall be Δ14°F above 100% power operating ambient temperature conditions as determined by DAEC Plant Test Procedure.

TABLE 3.2-C

Minimum No. of Operable Instrument Channels Per Trip System	Instrument	Trip Level Setting	Number of Instrument Channels Provided by Design	Action
2	APRM Upscale (Flow Biased)	$\leq (0.66\ W + 42)\ (\frac{FRP}{MFLPD})^{(2)}$	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)	$\leq (0.66\ W + 39)\ (\frac{FRP}{MFLPD})^{(2)}$	2 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)

NOTES FOR TABLE 3.2-C

1. For the startup and run positions of the Reactor Mode Selector Switch, there shall be two operable or tripped trip systems for each function. The SRM and IRM blocks need not be operable in "Run" mode, and the APRM [except for APRM Upscale (Not in Run Mode)] and RBM rod blocks need not be operable in "Startup" mode. If the first column cannot be met for one of the two trip systems, this condition may exist for up to seven days provided that during that time the operable system is functionally tested immediately and daily thereafter; if this condition lasts longer than seven days, the system shall be tripped. If the first column cannot be met for both trip systems, the systems shall be tripped.

2. W is the recirculation loop flow in percent of design. Trip level setting is in percent of rated power (1593 MWt). Refer to limiting Safety System Settings for variation with MFLPD & FRP (applicable only when MFLPD exceeds FRP).

3. IRM downscale is bypassed when it is on its lowest range.

4. This function is bypassed when the count rate is >100 cps.

4.5 BASES

Core and Containment Cooling Systems Surveillance Frequencies

The testing interval for the core and containment cooling systems is based on industry practice, quantitative reliability analysis, judgement and practicality. The core cooling systems have not been designed to be fully testable during operation. For example, in the case of the HPCI, automatic initiation during power operation would result in pumping cold water into the reactor vessel which is not desirable. Complete ADS testing during power operation causes an undesirable loss-of-coolant inventory. To increase the availability of the core and containment cooling systems, the components which make up the system; i.e., instrumentation, pumps, valves, etc., are tested frequently. The pumps and motor operated injection valves are also tested each month to assure their operability. A simulated automatic actuation test once each cycle combined with frequent tests of the pumps and injection valves is deemed to be adequate testing of these systems.

When components and subsystems are out-of-service, overall core and containment cooling reliability is maintained by demonstrating the operability of the remaining equipment. The degree of operability to be demonstrated depends on the nature of the reason for the out-of-service equipment. For routine out-of-service periods caused by preventative maintenance, etc., the pump and valve operability checks will be performed to demonstrate operability of the remaining components. However, if a failure due to a design deficiency, caused the outage, then the demonstration of operability should be thorough enough to assure that a generic problem does not exist. For example, if an out-of service period were caused by failure of a pump to deliver rated capacity due to a design deficiency, the other pumps of this type might be subjected to a flow rate test in addition to the operability checks.

Redundant operable components are subjected to increased testing during equipment out-of-service times. This adds further conservatism and increases assurance that adequate cooling is available should the need arise.

The RHR valve power bus is not instrumented. For this reason surveillance requirements require once per shift observation and verification of lights and instrumentation operability.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
	b. The indicated value of core flow rate varies from the value derived from loop flow measurements by more than 10%.
	c. The diffuser to lower plenum differential pressure reading on an individual jet pump varies from the mean of all jet pump differential pressures by more than 10%.
	2. Whenever there is recirculation flow with the reactor in the Start-up or Run mode, and one recirculation pump is operating, the diffuser to lower plenum differential pressure shall be checked daily and the differential pressure of an individual jet pump in loop shall not vary from the mean of all jet pump differential pressures in that loop by more than 10%.
F. Jet Pump Flow Mismatch	F. Jet Pump Flow Mismatch
1. When both recirculation pumps are in steady state operation, the speed of the faster pump may not exceed 122% of the speed of the slower pump when core power is 80% or more of rated power or 135% of the speed of the slower pump when core power is below 80% of rated power.	1. Recirculation pump speeds shall be checked and logged at least once per day.
2. From and after the date that one recirculation loop is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding 24 hours unless such loop is sooner made operable. If this requirement cannot be met, an orderly shutsown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours.	

LIMITING CONDITIONS FOR OPERATION

H. Shock Suppressors (Snubbers)

1. During all modes of operation, except Cold Shutdown and Refuel, all safety related snubbers listed in Tables 4.6-3 and 4.6-4 shall be operable, except as noted in 3.6.H.2 through 3.6.H.4 below.

2. From and after the time that a snubber is determined to be inoperable, continued reactor operation is permissible only during the succeeding 72 hours unless the snubber is sooner made operable or replaced.

3. If the requirements of 3.6.H.1 and 3.6.H.2 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 36 hours.

4. If a snubber is determined to be inoperable while the reactor is in the shutdown or refuel mode, the snubber shall be made operable or replaced prior to reactor startup.

5. Snubbers may be added to safety related systems without prior License Amendment to Tables 4.6-3 or 4.6-4 provided that a revision to Table 4.6-3 or 4.6-4 is included with the next License Amendment request.

SURVEILLANCE REQUIREMENTS

H. Shock Suppressors (Snubbers)

The following surveillance requirements apply to all hydraulic snubbers listed in Tables 4.6-3 and 4.6-4:

1. All hydraulic snubbers whose seal material has been demonstrated by operating experience, lab testing or analysis to be compatible with the operating environment shall be visually inspected. This inspection shall include, but not necessarily be limited to, inspection of the hydraulic fluid reservoir, fluid connections and linkage connections to the piping and anchor to verify snubber operability in accordance with the following schedule:

Number of Snubbers Found Inoperable During Inspection or During Inspection Interval	Next Required Inspection Interval
0	18 months ± 25%
1	12 months ± 25%
2	6 months ± 25%
3, 4	124 days ± 25%
5, 6, 7	62 days ± 25%
≥8	31 days ± 25%

The required inspection interval shall not be lengthened more than one step at a time.

Snubbers are categorized in two groups, "accessible and inaccessible," based on their accessibility for inspection during reactor operation. These two groups will be inspected independently according to the above schedule.

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LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
	2. All hydraulic snubbers whose seal materials are other than ethylene propylene or other material that has been demonstrated to be compatible with the operating environment shall be visually inspected for operability every 31 days.
	3. The initial inspection shall be performed within six months ± 25% from the date of issuance of these specifications. For the purpose of entering the schedule in Specification 4.6.H.1, it shall be assumed that the facility has been on a six-month inspection interval.
	4. Once each refueling cycle a representative sample of 10 hydraulic snubbers or approximately 10% of the hydraulic snubbers, whichever is less, shall be functionally tested for operability including verification of proper piston movement, lock-up and bleed. For each unit and subsequent unit found inoperable, an additional 10% or ten (10) hydraulic snubbers shall be so tested until no more failures are found or all units per category tested have been tested. Snubbers of rated capacity greater than 50,000 lbs. need not be functionally tested.

3.6.A & 4.6.A BASES:

Thermal and Pressurization Limitations

The thermal limitations for the reactor vessel meet the requirements of 10CFR50, Appendix G. 56

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 stesses were withim 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 FSAR Appendix K, "Field Fabricated 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

550°F range. The differential is due to the sluggish temperature response of the flange metal and its value decreases for any lower heating rate or the same rate applied over a narrower range.

The coolant in the bottom of the vessel is at a lower temperature than that in the upper regions of the vessel when there is no recirculation flow. This colder water is forced up when recirculation pumps are started. This will not result in stresses which exceed ASME Boiler and Pressure Vessel Code, Section III limits when the temperature differential is not greater than 145°F.

The reactor coolant system is a primary barrier against the release of fission products to the environs. In order to provide assurance that this barrier is maintained at a high degree of integrity, restrictions have been placed on the operating conditions to which it can be subjected.

The nil-ductility transition (NDT) temperature is defined as the temperature below which ferritic steel breaks in a brittle rather than ductile manner. Radiation exposure from fast neutrons ($>$1 mev) above about 10^{17} nvt may shift the NDT temperature of the vessel base metal above the initial value. Extensive tests have established the magnitude of changes as a function of the integrated neutron exposure.

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Neutron flux wires and samples of vessel material are installed in the reactor vessel adjacent to the vessel wall at the core midplane level. The wires and samples will be

removed and tested according to 10CFR50 Appendix H. Results of these analyses will be used to adjust Figure 3.6-1 as appropriate. 56

As described in paragraph 4.2.5 of the Safety Analysis report, detailed stress analyses have been made on the reactor vessel for both steady state and transient conditions with respect to material fatigue. The results of these transients are compared to allowable stress limits. Requiring the coolant temperature in an idle recirculation loop to be within 50°F of the operating loop temperature before a recirculation pump is started assures that the changes in coolant temperature at the reactor vessel nozzles and bottom head region are acceptable.

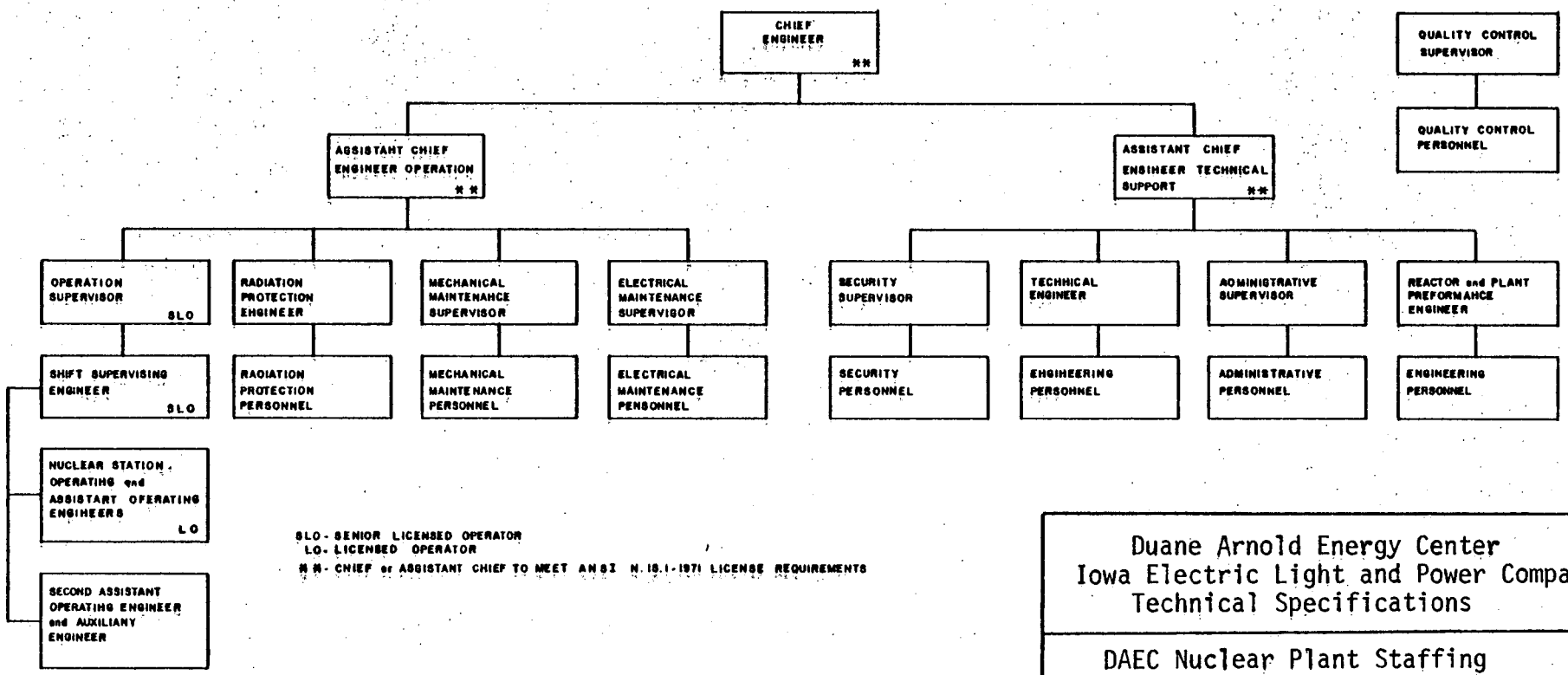
6.0 ADMINISTRATIVE CONTROLS

6.1 MANAGEMENT - AUTHORITY AND RESPONSIBILITY

6.1.1 The Chief Engineer has primary responsibility for the safe operation of the DAEC, and reports to the Assistant Vice President - Nuclear Generation.

6.1.2 The overall responsibility for the fire protection program for DAEC is assigned to the Assistant Vice President - Nuclear Generation. The DAEC Chief Engineer is responsible for directing the operating plant fire protection program.

6.1.3 The Quality Control Supervisor is responsible for implementation of the Quality Assurance Program at the DAEC and reports to the Manager - Quality Assurance.



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

DAEC Nuclear Plant Staffing
Figure 6.2-1

6.4 RETRAINING AND REPLACEMENT TRAINING

6.4.1 A training program shall be established to maintain the overall proficiency of the operating organization. This program shall consist of both retraining and replacement training elements and shall meet or exceed the minimum provisions outlined in ANSI N18.1-1971.

6.4.2 A training program for the fire brigade shall be maintained under the direction of the Chief Engineer and shall meet or exceed the requirements of Section 27 of the NFPA Code, except for fire brigade training sessions which shall be held at least quarterly.

6.5 REVIEW AND AUDIT

6.5.1 Operations Committee

6.5.1.1 Function

The Operations Committee shall function to advise the Chief Engineer on all matters related to nuclear safety.

6.5.1.2 Composition

The Operations Committee shall be composed of the Assistant Chief Engineers and Supervisors from the following departments: Operations, Maintenance, Reactor and Plant Engineering, Radiation Protection, Quality Control, and Technical Engineering.

The Assistant Chief Engineer - Operations shall act as the Chairman. One or more of the members shall be designated as Vice Chairman.

6.5.1.3 Alternates

All alternate members shall be appointed in writing by the Chief Engineer to serve on a permanent basis; however, no more than three alternates shall participate as voting members in Operations Committee activities at any one time.

6.6 REPORTABLE OCCURRENCE ACTION

6.6.1 Any reportable occurrence shall be reported immediately to the Chief Engineer and to the Assistant Vice President - Nuclear Generation, and promptly reviewed by the Operations Committee.

6.6.2 The Operations Committee shall prepare a separate report of each reportable occurrence. This report shall include an evaluation of the cause of the occurrence, a record of the corrective action taken, and also recommendation for appropriate action to prevent or reduce the probability of a recurrence.

6.6.3 Copies of all such reports shall be submitted to the Safety Committee for review and to the Assistant Vice President - Nuclear Generation for review and approval of any recommendations.

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 Assistant Vice President - Nuclear Generation and the Safety Committee. The Assistant Vice President - Nuclear Generation shall promptly report the circumstances to the NRC as specified in Subsection 6.12, 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 Assistant Vice President - Nuclear Generation and the Safety Committee. Appropriate analyses or reports will be submitted to the NRC by the Assistant Vice President - Nuclear Generation as specified in Subsection 6.12, Plant Reporting Requirements.

SPECIFICATION	SURVEILLANCE REQUIREMENT
6.9.2 Source Leakage Test	7.9.2 Source Leakage Test
A. Radioactive sources shall be leak tested for contamination. The leakage test shall be capable of detecting the presence of 0.005 microcurie of radioactive material on the test sample. If the test reveals the presence of 0.005 microcurie or more of removable contamination, it shall immediately be withdrawn from use, decontaminated, and repaired, or be disposed of in accordance with Commission regulations.	A. Tests for leakage and/or contamination shall be performed by the licensee or by other persons specifically authorized by the Commission or an agreement State, as follows:
Those quantities of byproduct material that exceed the quantities listed in 10 CFR 30.71 Schedule B are to be leak tested in accordance with the schedule shown in Surveillance Requirements. All other sources (including alpha emitters) containing greater than 0.1 microcurie are also to be leak tested in accordance with the Surveillance Requirements.	1. Each sealed source, except startup sources subject to core flux, containing radioactive material, other than Hydrogen 3, with a half-life greater than thirty days and in any form other than gas shall be tested for leakage and/or contamination at intervals not to exceed six months.
	2. The periodic leak test required does not apply to sealed sources that are stored and not being used. The sources excepted from this test shall be tested for leakage prior to any use or transfer to another user unless they have been leak tested within six months prior to the date of use or transfer. In the absence of a certificate from a transferor indicating that a test has been made within six months prior to the transfer, sealed sources shall not be put into use until tested.
	3. Startup sources shall be leak tested prior to and following any repair or maintenance and before being subjected to core flux.

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SPECIFICATION	SURVEILLANCE REQUIREMENT
B. Reporting Requirements Results of the leak tests performed on sources shall be included in the Annual Operating Report if the tests reveal the presence of 0.005 microcurie or more of removable contamination.	

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6.9.2 BASES

Ingestion or inhalation of source material may give rise to total body or organ irradiation. This specification assures that leakage from radioactive material sources does not exceed allowable limits. In the unlikely event that those quantities of radioactive by-product materials of interest to this specification which are exempt from leakage testing are ingested or inhaled, they represent less than one maximum permissible body burden for total body irradiation. The limits for all other sources (including alpha emitters) are based upon 10 CFR 70.39(c) limits for plutonium.