27. FREQUENCY NOTATION

NOTATION	FREQUENCY
S	At least once per 12 hours.
D	At least once per 24 hours.
W	At least once per 7 days.
M	At least once per 31 days.
Q	At least once per 92 days.
SA	At least once per 184 days.
A	At least once per year.
R	At least once per 18 months.
S/U	Prior to each reactor startup.
P	Prior to each release.
NA	Not applicable.

~~28. FIRE SUPPRESSION WATER SYSTEMS~~ 28. DELETED

~~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 alar device on each sprinkler, hose standpipe or deluge system riser.~~

29. REACTOR TRIP SYSTEM RESPONSE TIME

Reactor trip system response time is the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until deenergization of the scram pilot valve solenoids.

30. REPORTABLE EVENT

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

31. OFFSITE DOSE ASSESSMENT MANUAL

The Offsite Dose Assessment Manual (ODAM) contains the methodology and parameters to be used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints, and in the conduct of the Radiological Environmental Monitoring Program. The ODAM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Program required by Section 6.9.4 and (2) descriptions of the information that should be included in the Semiannual Radioactive Material Release Report and Annual Radiological Environmental Report required by the Technical Specification 6.11.1.

32. Deleted

33. PURGE - PURGING

PURGE or PURGING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

34. VENTING

VENTING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is not provided or required during the process. Vent, used in system names, does not imply a VENTING process.

35. PROCESS CONTROL PROGRAM (PCP)

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

36. MEMBER(S) OF THE PUBLIC

MEMBER(S) OF THE PUBLIC are persons who are not occupationally associated with Iowa Electric Light and Power Company and who do not normally frequent the DAEC site. The category does not include contractors, contractor employees, vendors or persons who enter the site to make deliveries or to service equipment.

37. SITE BOUNDARY

The Site Boundary is that line beyond which the land is neither owned, nor leased, nor otherwise controlled by IELP. UFSAR Figure 1.2-1 identifies the DAEC Site Boundary. For the purpose of implementing radiological effluent controls, the Unrestricted Area is that land (offsite) beyond the Site Boundary

38. ANNUAL

Occurring every 12 months.

For the purpose of designating surveillance test frequencies, annual surveillance tests are to be conducted at least once per 12 months.

39. CORE OPERATING LIMITS REPORT

The Core Operating Limits Report is the DAEC-specific document that provides cycle-specific operating limits for the current operating reload cycle. These cycle-specific operating limits shall be determined for each reload cycle in accordance with TS 6.11.2. Plant operation within these limits is addressed in individual technical specifications.

40. Shutdown Margin

Shutdown margin is the amount of reactivity by which the reactor is subcritical or would be subcritical assuming all control rods are inserted, except for the analytically strongest worth control rod, which is fully withdrawn, with the core in its most reactive state during the OPERATING ~~Cycle.~~ CYCLE.

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

At pressures below 785 psig, the core elevation pressure drop (0 power, 0 flow) is greater than 4.56 psi. At low power and all flows, this pressure differential is maintained in the bypass region of the core. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and all flows will always be greater than 4.56 psi. Analyses show that with a flow of 28 x 10^3 lbs/hr bundle flow, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.56 psi driving head will be greater than 28 x 10^3 lbs/hr irrespective of total core flow and independent of bundle power for the range of bundle powers of concern. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors this corresponds to a core thermal power of more than 50%. Thus, a core thermal power limit of 25% for reactor pressures below 800 psia or core flow less than 10% is conservative.

C. Power Transient

Plant safety analyses have shown that the scrams caused by exceeding any safety setting will assure that the Safety Limit of Specification 1.1.A or 1.1.B will not be exceeded. Scram times are checked periodically to assure the insertion times are adequate. The thermal power transient resulting when a scram is accomplished other than by the expected scram signal (e.g., scram from neutron flux folowing close of the main turbine stop valves) does not necessarily cause fuel damage. However, for this specification a Safety Limit violation will be assumed when a scram is only accomplished by means of a backup feature of the plant design. The concept of not approaching a Safety Limit provided scram signals are operable is supported by the extensive plant safety analysis.

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2. <u>APRM High Flux Scram (Refuel or Startup & Hot Standby Mode)</u>

For operation in these modes the APRM scram setting of 15 percent of rated power and the IRM High Flux Scram provide adequate thermal margin between the setpoint and the Safety Limit, 25 percent of rated. The margin is adequate to accommodate anticipated maneuvers associated with power plant startup. Effects of increasing pressure at zero or low void content are minor, cold water from sources available during startup is not much colder than that already in the system, temperature coefficients are small, and control rod patterns are constrained to be uniform by operating procedures backed up by the rod worth minimizer~~, and the Rod Sequence Control System.~~ Worths of individual rods are very low in a uniform rod pattern. Thus, of all possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power rise.

Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks, and because several rods must be moved to change power by a significant percentage of rated power, the rate of power rise is very slow. Generally, the heat flux is near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the scram level, the rate of power rise is no more than 5 percent of rated power per minute, and the APRM system would be more than adequate to assure a scram before the power could exceed the Safety Limit. The 15 percent

3. IRM

The IRM system consists of 6 chambers, 3 in each of the reactor protection system logic channels. The IRM is a 5-decade instrument which covers the range of power level between that covered by the SRM and the APRM. The 5 decades are covered by the IRM by means of a range switch and the 5 decades are broken down into 10 ranges, each being one-half of a decade in size. The IRM scram trip setting of 120 divisions is active in each range of the IRM. For example, if the instrument were on range 5, the scram would be 120 divisions on that range. Thus, as the IRM is ranged up to accommodate the increase in power level, the scram trip setting is also ranged up. The most significant sources of reactivity change during the power increase are due to control rod withdrawal. For insequence control rod withdrawal, the rate of change of power is slow enough due to the physical limitation of withdrawing control rods that the heat flux is in equilibrium with the neutron flux, and the IRM scram would result in a reactor shutdown well before any Safety Limit is exceeded.

In order to ensure that the IRM provides adequate protection against the single rod withdrawal error, a range of rod withdrawal accidents has been analyzed. This analysis included starting the accident at various power levels. The most severe case involves an initial condition in which the reactor is just subcritical and the IRM system is not yet on scale. This condition exists at quarter rod density. Additional conservatism was taken in this analysis by assuming that

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING
	C. Relief Valve Settings - Low-Low Set Function

Valve Group	Open	Close
Low (1 valve)	1020 psig	900 psig
High (1 valve)	1025 psig	905 psig.

All settings are ± 25 psi.

D. Safety Valve settings

1240 psig ± 12 psi
(2 valves)

2. The shutdown cooling isolation valves shall be closed whenever the reactor vessel dome pressure is ≥ 135 psig.

2. ~~The reactor~~ vessel dome presure shall not exceed 135 psig at any time when operating the Residual Heat Removal pump in the shutdown cooling mode.

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LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
d. Each control rod shall be coupled to its drive.* If a control rod becomes uncoupled, (i) recouple the control rod within 2 hours and (ii) verify coupling by performing surveillance 4.3.A.2.d. (iii) If the control rod is not recoupled, declare the control rod inoperable. The actions stated in Specification 3.3.A.2.e shall be taken.	d. When a control rod is withdrawn the first time after refueling, after CRD maintenance or when required by Specification 3.3.A.2.d(ii), coupling integrity shall be verified by observing that the drive does not go to the overtravel position when the rod is fully withdrawn.
e. A control rod that has been declared inoperable for reasons other than being stuck shall: (i) be fully inserted,** and (ii) disarm the associated directional control valves electrically. The control valves may be re-armed to permit testing associated with returning the control rod to OPERABLE status. (iii) Whenever the reactor is less than 20% power, verify all inoperable control rods not in compliance with BPWS are separated by 2 or more OPERABLE control rods in any direction, including the diagonal. (iv) Verify that no more than 8 inoperable control rods exist. (v) If the requirements of Specification 3.3.A.2.e (i)-(iv) cannot be met, be in COLD SHUTDOWN within 24 hours.	e. (not used)

* This requirement does not apply in the refuel condition. ~~Refer to~~ Specifications 3.9.A.~~5~~ 3 and 3.9.A.~~6~~ 4 for control rod requirements during refueling.

**The RWM may be bypassed, if required, to allow insertion of inoperable control rods and continued operation.

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LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
C. Residual Heat Removal (RHR) Service Water System	C. Surveillance of the RHR Service Water System
1. Except as specified in 3.5.C.2, 3.5.C.3, 3.5.C.4, ~~3.5.C.5,~~ and 3.5.G.3 below, both RHR service water subsystem loops shall be operable whenever irradiated fuel is in the reactor vessel and reactor coolant temperature is greater than 212°F.	1. Surveillance of the RHR service water system shall be as follows: RHR Service Water Subsystem Testing: **Item** / **Frequency** a. Pump and Motor operated valve operability. — Once/3 months b. Flow Rate Test-Each RHR service water pump shall deliver at least 2040 gpm at a TDH of 610 ft. or more. — after major pump maintenance and every 3 months
2. With one RHRSW pump inoperable, provided the remaining active components of both RHRSW subsystems are verified to be OPERABLE, restore the inoperable pump to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.	
3. With one RHRSW pump in each subsystem inoperable, provided the remaining active components of both RHRSW subsystems and the diesel generators are verified to be OPERABLE, restore at least one inoperable pump to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.	

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LIMITING CONDITION FOR OPERATION | SURVEILLANCE REQUIREMENT

4. With one RHRSW subsystem inoperable, provided the remaining RHRSW subsystem and its associated diesel generator are verified to be OPERABLE, restore the inoperable system to OPERABLE status with at least one OPERABLE pump within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

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.~~

D. HPCI Subsystem

1. HPCI Subsystem testing shall be performed as follows:

Item	Frequency
a. Simulated Automatic Actuation Test	Annual
b. Pump Operability	Once/3 Months
c. Motor Operated Valve Operability	Once/3 Months
d. At rated reactor pressure demonstrate ability to deliver rated flow at a discharge pressure greater than or equal to that pressure required to accomplish vessel injection if vessel pressure were as high as 1040 psig.	Once/3 months

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3.6.F & 4.6.F BASES:

Jet Pump Flow Mismatch

The LPCI loop selection logic has been previously described in the Updated FSAR Section 7.3.1.1.2.4. ~~7.3.1.1.24~~. For some limited low probability accidents with the recirculation loop operating with large speed differences, it is possible for the logic to select the wrong loop for injection. For these limited conditions the core spray itself is adequate to prevent fuel temperatures from exceeding allowable limits. However, to limit the probability even further, a procedural limitation has been placed on the allowable variation in speed between the recirculation pumps.

The licensee's analyses indicate that above 80% power the loop select logic could be expected to function at a speed differential up to 14% of their average speed. Below 80% power the loop select logic would be expected to function at a speed differential up to 20% of their average speed. This specification provides margin because the limits are set at ±10% and ±15% of the average speed for the above and below 80% power cases, respectively. If the reactor is operating on one recirculation pump, the loop select logic trips that pump before making the loop selection.

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The pressure suppression pool water provides the heat sink for the reactor primary system energy release following a postulated rupture of the system. The pressure suppression chamber water volume must absorb the associated decay and structural sensible heat released during primary system blowdown from 1040 psig. Since all of the gases in the drywell are purged into the pressure suppression chamber air space during a loss-of-coolant accident, the pressure resulting from isothermal compression plus the vapor pressure of the liquid must not exceed 62 psig, the suppression chamber maximum allowable pressure. The design volume of the suppression chamber (water and air) was obtained by considering that the total volume of reactor coolant to be condensed is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber.

Using the minimum or maximum water volumes given in the specification, containment pressure during the design basis accident is approximately 43 psig which is below the design pressure of 56 psig. The minimum volume of 58,900 ft^3 results in a submergence of approximately 3 feet. Based on Humboldt Bay, Bodega Bay, and Marviken test facility data as utilized in General Electric Company document number NEDE-21885-P and data presented in Nutech document, Iowa Electric document number 7884-M325-002, the following technical assessment results were arrived at:

1. Condensation effectiveness of the suppression pool can be maintained for both short and long term phases of the Design Basis Accident (DBA), Intermediate Break Accident (IBA), and Small Break Accident (SBA) cases with three feet submergence.

8. Procedures required by the plant Security Plan.

9. Operation of radioactive waste systems.

10. Fire Protection Program implementation.

11. A preventive maintenance and periodic visual examination program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient to as low as practical levels. This program shall also include provisions for performance of periodic systems leak tests of each system once per operating cycle.

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

13. Administrative procedures for shift overtime for Operations personnel to be consistent with the Commission's June 15, 1982 policy statement.

14. Offsite Dose Assessment Manual.

15. Process Control ~~Plan.~~ Program.

16. Quality Control Program for effluents.

6.8.2 Procedures described in 6.8.1 above, and changes thereto, shall be reviewed by the Operations Committee as indicated in Specification 6.5.1.6 and approved by the Plant Superintendent-Nuclear prior to implementation, except as provided in 6.8.3 below.

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

(5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.

6.11.2 CORE OPERATING LIMITS REPORT

a. Core cycle-dependent limits shall be established prior to each reload cycle, or prior to any remaining part of a reload cycle, for the following:

1) Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) - Specification 3.12.A.

2) Linear Heat Generation Rate (LHGR) - Specification 3.12.B.

3) Minimum Critical Power Ratio (MCPR) - Specification 3.12.C.

4) $MAPFAC_f$ and $MAPFAC_p$ Factors which multiply the MAPLHGR limits - Specification 3.3.F.4.a.

These limits shall be documented in the CORE OPERATING LIMITS REPORT.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC in General Electric Standard Application for Reactor Fuel, NEDE-24011-P-A, (GESTAR II).*

c. The core operating limits shall be determined such that all applicable limits (e.g. fuel thermal-mechanical limits, core thermal hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

d. The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

6.11.3 UNIQUE REPORTING REQUIREMENTS

Special reports shall be submitted to the Director of Inspection and Enforcement Regional Office within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification.

a. Reactor vessel base, weld and heat affected zone metal test specimens (Specification 4.6.A.2).

b. deleted

c. Inservice inspection (Specification 4.6.G.).

d. Reactor Containment Integrated Leakage Rate Test (Specification 4.7.A.2.f).

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*Approved revision number at time reload fuel analyses are performed.

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e. deleted

f. deleted

g. deleted

h. Radioactive Liquid or Gaseous Effluent - calculated dose exceeding specified limit (ODAM Sections 6.1.3, 6.2.3 and 6.2.4).

i. Off-Gas System inoperable (ODAM Section 6.2.5).

j. Measured levels of radioactivity in an environmental sampling medium determined to exceed the reporting level values of ODAM Table 6.3-3 when averaged over any calendar quarter sampling period (ODAM Section ~~7.3.2.2~~ 6.3.2.1).

k. Annual dose to a MEMBER OF THE PUBLIC determined to exceed 40 CFR Part 190 dose limit (ODAM Section 6.3.1.1).

l. Radioactive liquid waste released without treatment when activity concentration ~~exceeds~~ is equal to or greater than 0.01 μci/ml (ODAM Section 6.1.4.1).

m. Explosive Gas Monitoring Instrumentation Inoperable (Specification 3.2.I.1).

n. Liquid Holdup Tank Instrumentation Inoperable (Specification 3.14.B.1).

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SAFETY ASSESSMENT

Introduction

By letter dated December 22, 1993, Iowa Electric Light and Power Company (IELP) submitted a request for revision of the Technical Specifications (TS) for the Duane Arnold Energy Center (DAEC). The proposed changes are editorial in nature and will eliminate discrepancies discovered in the recent comparison of the IELP controlled copy with the NRC authority copy of the DAEC TS.

Assessment

The proposed changes to the DAEC TS correct administrative and typographical errors. None of the revisions would result in a change to the systems, structures or components in the plant or operation thereof. The revised TS will result in continued safe operation of the DAEC and elimination of some administrative and typographical errors in the TS.

Based upon the above information, we have concluded that the proposed changes to the DAEC TS are acceptable.

ENVIRONMENTAL CONSIDERATION

10 CFR 51.22(c)(9) identifies certain licensing and regulatory actions which are eligible for categorical exclusion from the requirement to perform an environmental assessment. A proposed amendment to an operating license for a facility requires no environmental assessment if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant hazards consideration; (2) result in a significant change in the types or significant increase in the amounts of any effluents that may be released offsite; and (3) result in an increase in individual or cumulative occupational radiation exposure. Iowa Electric Light and Power has reviewed this request and determined that the proposed amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with the issuance of the amendment. The basis for this determination follows:

Basis

The change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) for the following reasons:

1. As demonstrated in Attachment 1 to this letter, the proposed Amendment does not involve a significant hazards consideration.

2. The proposed Amendment makes editorial changes to correct administrative and typographical errors. No changes in either design or operation of the DAEC will be made as a result of these changes; thus, there will be no increase in either the types or amounts of effluents that may be released offsite.

3. The proposed Amendment makes editorial changes to correct administrative and typographical errors. No changes in either design or operation of the DAEC will be made as a result of these changes; thus, there will be no significant increase in either individual or cumulative occupational exposure.