UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

IOWA ELECTRIC LIGHT AND POWER COMPANY
CENTRAL IOWA POWER COOPERATIVE
CORN BELT POWER COOPERATIVE

DOCKET NO. 50-331

DUANE ARNOLD ENERGY CENTER

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 180
License No. DPR-49

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Iowa Electric Light and Power Company, et al., dated July 6, 1990, revised August 30, 1991, and January 8, 1992, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. DPR-49 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 180, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. The license amendment is effective as of the date of issuance and shall be implemented within 60 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Clyde Y. Shiraki, Sr. Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of issuance: March 11, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 180

FACILITY OPERATING LICENSE NO. DPR-49

DOCKET NO. 50-331

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised areas are indicated by marginal lines.

Pages

i

vii

1.0-9

3.2-2

3.2-2a

3.3-1

3.3-2

3.3-3

3.3-4

3.3-5

3.3-6

3.3-7

3.3-8

3.3-9

3.3-10

3.3-11

3.3-12

3.3-13

3.3-14

3.3-15

3.9-2

TABLE OF CONTENTS

TECHNICAL SPECIFICATIONS

LIST OF FIGURES

Mr. Lee Liu
Iowa Electric Light and Power Company   Duane Arnold Energy Center

cc:
Jack Newman, Esquire
Kathleen H. Shea, Esquire
Newman and Holtzinger
1615 L Street, N.W.
Washington, D.C. 20036

Chairman, Linn County
Board of Supervisors
Cedar Rapids, Iowa 52406

Iowa Electric Light and Power Company
ATTN: David L. Wilson
Post Office Box 351
Cedar Rapids, Iowa 52406

U.S. Nuclear Regulatory Commission
Resident Inspector's Office
Rural Route #1
Palo, Iowa 52324

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
799 Roosevelt Road
Glen Ellyn, Illinois 60137

Mr. Stephen N. Brown
Utilities Division
Iowa Department of Commerce
Lucas Office Building, 5th Floor
Des Moines, Iowa 50319

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 generally describe the essential process controls and checks used to assure that a process for solidifying radioactive waste from a liquid system produces a product that is acceptable for burial according to 10 CFR Part 61.56.

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 technical specifications, 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.

LIMITING CONDITIONS FOR OPERATION

C. Control Rod Block Actuation

1. SRM, IRM, APRM and Scram Discharge Volume Rod Blocks

The Limiting Conditions of Operation for the instrumentation that initiates these control rod block are given in Table 3.2-C.

2. Rod Block Monitor (RBM)

(a) The RBM control rod block setpoints are given in Table 3.2-C. The upscale High Power Trip Setpoint shall be applied when the core thermal power is greater than or equal to 85% of rated ($P \geq 85\%$). The upscale Intermediate Power Trip Setpoint shall be applied when the core thermal power is greater than or equal to 65% of rated and less than 85% of rated ($65\% \leq P < 85\%$). The upscale Low Power Trip Setpoint shall be applied when the core thermal power is greater than or equal to 30% of rated and less than 65% of rated ($30\% \leq P < 65\%$). The RBM can be bypassed when core thermal power is less than 30% of rated. The RBM bypass time delay (t_{d2}) shall be less than or equal to 2.0 seconds.

SURVEILLANCE REQUIREMENT

C. Control Rod Block Actuation

1. Instrumentation shall be functionally tested, calibrated and checked as indicated in Table 4.2-C.

System logic shall be functionally tested as indicated in Table 4.2-C.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT
D. Radiation Monitoring Systems-Isolation & Initiation Functions	D. Radiation Monitoring Systems-Isolation & Initiation Functions
1. Steam Air Ejector Offgas System	1. Steam Air Ejector Offgas System
a) At least one post-treatment steam air ejector offgas system radiation monitor shall be operable during reactor power operation. The monitors shall be set to initiate immediate closure of the charcoal bed bypass valve and the air ejector offgas isolation valve at a setting equivalent to or below the dose rate limits in Specification 3.15.2.1.	Instrumentation shall be functionally tested, calibrated and checked as indicated in Table 4.2.D. System logic shall be functionally tested as indicated in Table 4.2-D.
b) In the event no post-treatment monitor is operable, gases from the steam air ejector offgas system may be released to the environment for up to 72 hours provided (1) the charcoal bed of the offgas system is not bypassed, and (2) the offgas stack noble gas activity monitor is operable. Otherwise be in at least HOT STANDBY within the following 24 hours.	

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
3.3 REACTIVITY CONTROL	4.3 REACTIVITY CONTROL
Applicability:	Applicability:
Applies to the operational status of the control rod system.	Applies to the surveillance requirements of the control rod system.
Objective:	Objective:
To assure the ability of the control rod system to control reactivity.	To verify the ability of the control rod system to control reactivity.
Specification:	Specification:
A. Reactivity Limitations	A. Reactivity Limitations
1. Reactivity margin - core loading	1. Reactivity margin - core loading
A sufficient number of control rods shall be OPERABLE such that a SHUTDOWN MARGIN of at least 0.38% $\Delta k/k$ exists or be in COLD SHUTDOWN within 24 hours.	Prior to or during the first startup following CORE ALTERATIONS, verify that the required SHUTDOWN MARGIN exists by measurement during control rod withdrawal.
2. Reactivity margin - inoperable control rods	2. Reactivity margin - inoperable control rods
a. If one control rod scram accumulator is inoperable, (i) verify reactor Pressure is greater than 950 psig and (ii) restore the accumulator to OPERABLE status within 8 hours (iii) If the requirements of Specification 3.3.A.2.a (i) or (ii) cannot be met or greater than one accumulator is inoperable, the control rod(s) shall be declared inoperable and the actions stated in Specification 3.3.A.2.e shall be taken.	a. At least once per week, during Reactor Power Operation, verify the pressure and level alarms for each OPERABLE scram accumulator are not in the alarmed condition.
b. If a control rod(s) position cannot be determined, declare the rod inoperable. The actions stated in Specification 3.3.A.2.e shall be taken.	b. At least once per 24 hours, determine the position of each control rod.
c. Control rods with scram times greater than those permitted by Specification 3.3.D.3 shall be declared inoperable.	c. (not used)

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 and 3.9.A.6 for control rod requirements during refueling.

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

LIMITING CONDITION FOR OPERATION

f. A control rod which is not moveable with drive or scram pressure (stuck) shall be declared inoperable and the following actions shall be taken.

(i) Disarm the associated control rod drive and

(ii) verify compliance with Specification 3.3.A.1.

(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) within 48 hours, verify that the cause of the failure is not due to a failed control rod drive mechanism collet housing.

(v) if the requirements of Specification 3.3.A.2.f (i)-(iv) cannot be met or more than one control rod is stuck, be in COLD SHUTDOWN within 24 hours.

3. Control Rod Drive Housing Support

The control rod drive housing support system shall be in place whenever the reactor vessel is pressurized above atmospheric pressure with fuel in the reactor vessel.

B. Scram Discharge Volume

(Not Used)

SURVEILLANCE REQUIREMENT

f. Whenever the reactor is operating greater than 20% power:

(i) each partially or fully withdrawn operable control rod shall be demonstrated to be moveable by exercising it one notch at least once per week.

(ii) if a control rod cannot be moved with drive or scram pressure, each partially or fully withdrawn OPERABLE control rod shall be exercised one notch at least once each 24 hours, unless it has been determined that the failure is not a failed control rod drive mechanism collet housing.

3. Control Rod Drive Housing Support

The control rod drive housing support system shall be inspected after reassembly and the results of the inspection recorded.

B. Scram Discharge Volume

1. At least once per month, verify the SDV vent and drain valves are open.

2. At least once per quarter verify that

a. the SDV vent and drain valves close within 30 seconds after receipt of

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
	a close signal and
	b. after removal of the close signal, the vent and drain valves are open.
	3. At least once per OPERATING CYCLE, verify that
	a. the SDV vent and drain valves close within 30 seconds after receipt of a signal for the control rods to scram and
	b. the SDV vent and drain valves open when the scram signal is reset.
C. <u>REACTIVITY CONTROL SYSTEMS</u>	C. <u>REACTIVITY CONTROL SYSTEMS</u>
1. Whenever the reactor is operating at less than 20% power, the Rod Worth Minimizer (RWM) shall be OPERABLE or	1. Prior to the start of control rod withdrawal towards criticality and prior to obtaining 20% RATED POWER during rod insertion at shutdown, the capability of the RWM shall be verified by the following checks.
a. With the RWM inoperable after the first 12 rods are fully withdrawn, operation may continue provided that a second licensed operator verifies control rod movement and compliance with the prescribed control rod pattern.	a. The correctness of the Banked Position Withdrawal Sequence (or equivalent) input to the RWM computer shall be verified.
b. With the RWM inoperable before the first 12 control rods are fully withdrawn, one startup per calendar year may be performed provided that a second licensed operator verifies control rod movement and compliance with the prescribed control rod pattern.	b. The RWM computer on-line diagnostic test shall be successfully performed.
	c. Proper annunciation of the selection error of at least one out-of-sequence control rod in each fully inserted group shall be verified.
c. Otherwise, with the RWM inoperable, control rod movement shall not be permitted except by a scram.	d. The rod block function of the RWM shall be verified by demonstrating the inability to withdraw an out-of-sequence control rod.
2. Control rods shall not be withdrawn in STARTUP or REFUEL modes unless at least two Source Range Monitor Channels have an observed count rate equal to or greater than three counts per second.	2. Prior to control rod withdrawal in STARTUP or REFUEL modes, verify that at least two Source Range Monitor Channels have an observed count rate of at least three counts per second.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

3. During operation with Limiting Control Rod Patterns, either:

a. both RBM channels shall be OPERABLE, or

b. with one RBM channel inoperable, control rod withdrawal shall be blocked within 24 hours, unless OPERABILITY is restored within this time period, or

c. with both RBM channels inoperable, control rod withdrawal shall be blocked until OPERABILITY of at least one channel is restored.

D. Scram Insertion Times

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

Rod Position	Average Scram Insertion Times (Sec)
46	0.35
38	0.937
26	1.86
06	3.41

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

Rod Position	Average Scram Insertion Times (Sec)
46	0.37
38	1.01
26	1.97
06	3.62

3. Maximum scram insertion time to rod position 04 of any OPERABLE control rod should not exceed 7.00 seconds.

3. When a Limiting Control Rod Pattern exists and one RBM channel is inoperable, an Instrument Functional Test of the operable RBM channel shall be performed within 24 hours prior to rod withdrawal.

D. Scram Insertion Times

1. After each refueling outage all OPERABLE rods shall be scram time tested from the fully withdrawn position to the drop-out of the reed switch at the rod position required by Specification 3.3.D. The nuclear system pressure shall be above 950 psig (with saturation temperature). This testing shall be completed prior to exceeding 40% power. During all scram time testing below 20% power, the Rod Worth Minimizer shall be OPERABLE or a second licensed operator shall verify that the operator at the reactor console is following the control rod program.

LIMITING CONDITION FOR OPERATION

4. If Specification 3.3.D.1, 2 or 3 cannot be met, be in COLD SHUTDOWN within 24 hours.

E. Reactivity Anomalies

The reactivity difference between the actual rod density and predicted rod density shall not exceed 1% $\Delta k/k$.

1. If the reactivity is different by more than 1% $\Delta k/k$, perform an analysis to determine and explain the cause of the reactivity difference; operation may continue if the difference is explained and corrected.

2. Otherwise be in COLD SHUTDOWN within 24 hours.

F. Recirculation Pumps

When the reactor mode switch is in STARTUP or RUN position, the reactor shall not be operated in the natural circulation flow mode.

With two recirculation pumps in operation and with core thermal power greater than the limit specified in Figure 3.3-1 and total core flow less than 45% of rated, the APRM and LPRM* neutron flux noise levels shall be determined within 2 hours, and:

1. if the APRM and LPRM* neutron flux noise levels are less than or equal to three times their established baseline levels, continue to determine the noise levels at least once per 8 hours and also within 30 minutes after the completion of a core thermal power increase of at least 5% of rated core thermal power while operating in this region of the power/flow map, or

2. if the APRM and/or LPRM* neutron flux noise levels are greater than three times their established baseline levels, immediately initiate corrective action and restore the noise levels to within the required limits within 2 hours by increasing core flow, and/or by initiating an orderly reduction of

SURVEILLANCE REQUIREMENT

E. Reactivity Anomalies

The rod density shall be predicted and compared to the actual rod density:

1. during the first startup following CORE ALTERATIONS and

2. at least once per full power month.

F. Recirculation Pumps

With two recirculation pumps in operation and with core thermal power greater than the limit specified in Figure 3.3-1 and total core flow less than 45% of rated, establish baseline APRM and LPRM* neutron flux noise levels within 2 hours, provided that baseline values have not been previously established since the last core refueling.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
core thermal power by inserting control rods. See Specifications 3.6.F.2 for SLO. A recirculation pump shall not be started while the reactor is in natural circulation flow and reactor power is greater than 1% of RATED POWER. *Detector levels A and C of one LPRM string per core octant plus detector levels A and C of one LPRM string in the center of the core shall be monitored.	

3.3 and 4.3 BASES

A. Reactivity Limitation

1. Reactivity Margin - Core Loading

The requirements for the control rod drive system have been identified by evaluating the need for reactivity control via control rod movement over the full spectrum of plant conditions and events. As discussed in Subsection 4.6.1 of the Updated FSAR, the control rod system design is intended to provide sufficient control of core reactivity that the core could be made subcritical with the strongest rod fully withdrawn. This reactivity characteristic has been a basic assumption in the analysis of plant performance. Verification of this required shutdown margin is performed prior to or during the first startup after core alterations by measurement during control rod withdrawal. This demonstration is performed with the reactor core in the cold, xenon-free condition and will show that the reactor is subcritical by at least R + 0.38% $\Delta k/k$ with the analytically determined strongest control rod fully withdrawn. (Shutdown margin can also be verified, when actual demonstration is not feasible, by an analytical determination of the highest rod worth and core reactivity.)

The value of "R", in units of % $\Delta k/k$, is the amount by which the core reactivity, in the most reactive condition at any time in the subsequent operating cycle, is calculated to be greater than at the time of the demonstration. "R", therefore, is the difference between the calculated value of maximum core reactivity during the operating cycle and the calculated beginning-of-life core reactivity. The value of "R" must be positive or zero and must be determined for each fuel cycle.

In determining the "analytically strongest" rod, it is assumed that every fuel assembly of the same type has identical material properties. In the actual core, however, the control cell material properties vary within allowed manufacturing tolerances, and the strongest rod is determined by a combination of the control cell geometry and local $k\infty$. Therefore, an additional margin is included in the shutdown margin test to account for the fact that the "analytically strongest" rod is not necessarily the strongest rod in the core. Studies have been made which compare experimental criticals with calculated criticals. These studies have shown that actual criticals can be predicted within a given tolerance band. For gadolinia cores the additional margin required due to control cell material manufacturing tolerances and calculational uncertainties has experimentally been determined to be 0.38% $\Delta k/k$. When this additional margin is demonstrated, it assures that the reactivity control requirement is met.

2. Reactivity margin - inoperable control rods

Control rod operability (the capability to insert the control rods) ensures that the assumptions for scram reactivity in the safety analyses are not violated. Operability of an individual control rod is based on a combination of factors, primarily the scram insertion times, control rod scram accumulator status, control rod coupling integrity, control rod movability and the ability to determine a control rod's position. When a rod is declared inoperable, strict control over the number and distribution of inoperable control rods is required for the assumptions of the safety analyses and to provide early indication of potential generic problems in the CRD system.

The control rod drive scram accumulators are part of the CRD system and are provided to ensure adequate control rod scram under varying reactor conditions. The safety analyses assume that all of the control rods scram at a specified insertion rate. Surveillance of the control rod scram accumulator provides assurance (along with the other surveillances of control rod operability and scram insertion times) that the scram reactivity assumed in the safety analyses can be met. Because of the large number of control rods available for scram and the assumed single failure of a control rod to scram in the Safety Analysis, a specified amount of time is allowed to restore the accumulator to OPERABLE status. This time is only allowed, however, if reactor pressure is sufficient to scram the rod without help of the accumulator. If the accumulator cannot be restored to OPERABLE status, the associated control rod could potentially have a degraded scram insertion speed and therefore must be declared inoperable.

Control rod position information is required to ensure adequate information is available to the operator for determining CRD operability and controlling rod patterns. However, if a rod's position is not displayed, a control rod's position can be determined by moving the control rod to a position with an OPERABLE indicator or by the use of other appropriate methods.

Control rod drop accidents as discussed in the Updated FSAR can lead to significant core damage. If coupling integrity is maintained, the possibility of a rod drop accident is eliminated. The overtravel position feature provides a positive check as only uncoupled drives may reach this position. Continued operation with an uncoupled control rod should not be allowed because of the increased probability of a CRDA. Therefore, only a short period of time is allowed to establish and verify coupling. Since the allowable time with an uncoupled control rod is short, and control rods do not always couple on the first try, multiple attempts to recouple a control rod may be necessary.

Specification 3.3.A.2.e requires that a rod be taken out of service if it is declared inoperable. It is required to be fully inserted and disarmed electrically* to ensure it is in a safe position of maximum contribution to shutdown margin and to prevent inadvertent withdrawal during subsequent operations. Consideration of the control rod drop accident (CRDA) requires that the inoperable inserted control rods not

*To disarm the drive electrically, four Amphenol type plug connectors are removed from the drive insert and withdrawal solenoids rendering the rod incapable of withdrawal. This procedure is equivalent to valving out the drive and is preferred because, in this condition, drive water cools and minimizes crud accumulation in the drive. Electrical disarming does not eliminate position indication.

in compliance with BPWS (out-of-sequence) be separated by at least two operable control rods in any direction including the diagonal and that no more than 8 control rods are declared inoperable. Controlling the distribution of out-of-sequence control rods limits the potential reactivity worth of adjacent control rods. Limiting the number of inoperable rods to less than or equal to 8 ensures that possible generic problems are investigated and resolved.

If a control rod is declared inoperable and cannot be fully inserted (stuck), its position must be verified to be in compliance with the required SDM. This assures that the core can be shutdown at all times with the remaining control rods assuming the strongest operable control rod does not insert. In addition, the control rod should be isolated from a scram to protect the CRD and surrounding fuel assemblies should a scram occur. The CRD can be isolated from scram by isolating the hydraulic control unit from scram and normal insert/withdraw pressure yet still maintain cooling water to the CRD. Once this is done, the accumulator should be depressurized.

If the control rod is immovable and damage within the control rod drive mechanism and, in particular, cracks in the drive internal housings cannot be ruled out, then a generic problem affecting a number of drives cannot be ruled out. Circumferential cracks resulting from stress assisted intergranular corrosion have occurred in the collet housing of drives at several BWRs. This type of cracking could occur in a number of drives and if the cracks propagated until severance of the collet housing occurred, scram could be prevented in the affected rods. Limiting the period of operation with a potentially severed collet housing and requiring increased surveillance after detecting one stuck rod will assure that the reactor will not be operated with a large number of rods with failed collet housings.

3. Control Rod Drive Housing Support

The control rod housing support restricts the outward movement of a control rod to less than 3 inches in the extremely remote event of a housing failure. The amount of reactivity which could be added by this small amount of rod withdrawal, which is less than a normal single withdrawal increment, will not contribute to any damage to the primary coolant system. The design basis is given in Subsection 4.6.1 of the Updated FSAR and the safety evaluation is given in Subsection 4.6.2 of the Updated FSAR. This support is not required if the reactor coolant system is at atmospheric pressure since there would then be no driving force to rapidly eject a drive housing.

B. Scram Discharge Volume

1. To ensure that a volume exists to accept discharge water from the control rods during a reactor scram, the scram discharge volume (SDV) vent and drain valves are required to undergo surveillance testing. For the

monthly verification of SDV vent and drain valve status, observation of control room indicating lights is acceptable.

C. Reactivity Control Systems

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

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

   These control rod patterns are in accordance with the Banked Position Withdrawal Sequence (BPWS) (Ref. 1). The BPWS has the advantage of having been proven statistically to have such low individual control rod worths that the possibility of a control rod drop accident (CRDA), which exceeds the 280 cal/gm peak fuel enthalpy limit, is precluded (Ref. 2). The Reduced Notch Worth Procedure (RNWP) may be used in place of BPWS because the RNWP is an extension of BPWS (Ref. 3).

   In performing the function described above, the RWM is not required to impose any restrictions at core power levels in excess of 10% of rated. Material in the cited references shows that it is impossible to reach 280 cal/gm in the event of a control rod drop occurring at power greater than 10%, regardless of the rod pattern. This is true for all normal and abnormal patterns including those which maximize the individual rod worth.

   Power level for automatic cutout of the RWM function is sensed by feedwater and steam flow and is set nominally at 20% of rated power to account for instrument error.

   The RWM provides automatic supervision to assure that out-of-sequence control rods will not be withdrawn or inserted; i.e., it limits operator deviations from planned withdrawal sequences. The RWM serves as a backup to procedural control of control rod sequences, which limit the maximum reactivity worth of control rods. In the event that the Rod Worth Minimizer is out of service at less than 20% of rated power, a second Licensed Operator or other qualified technical plant employee whose qualifications have been reviewed by the NRC shall verify control rod movement and compliance with the prescribed control rod pattern.

   The functions of the RWM make it unnecessary to specify a license limit on rod worth to preclude unacceptable consequences in the event of a CRDA. At low powers, below 20%, this device forces adherence to acceptable rod patterns. Above 20% of rated power, no constraint on rod pattern is required to assure that the consequences of a CRDA are acceptable.

Functional testing of the RWM prior to the start of control rod withdrawal at startup, and prior to attaining 20% rated thermal power during rod insertion while shutting down, will ensure reliable operation.

If a rod is declared inoperable, adherence to the BPWS (and RNWP) is maintained by performing those actions required for an inoperable rod. These actions require fully inserting the inoperable rod, disarming it electrically, ensuring it is separated from other inoperable rods by at least two operable rods in any direction (if it is out-of-sequence) and having a maximum of 8 inoperable rods.

The operability requirements for the RWM have been established to minimize reactor operations without the RWM. If the operability requirements of the RWM are not satisfied, i.e., RWM is inoperable without the second licensed operator or the BPWS (RNWP) requirements for inoperable rods are not met (below 20% rated), then further rod movement is not permitted, except by a scram (manual scram or mode switch to SHUTDOWN). This is done to ensure that high rod worths, with the potential to exceed 280 cal/gm during a CRDA are not generated. However, limited rod movement shall be permitted solely for the purpose of troubleshooting and/or testing the RWM for OPERABILITY. Limited rod movement is defined as the movement of control rod(s) only to the extent necessary to determine that the rod inhibit functions of the RWM are working properly.

2. The Source Range Monitor (SRM) system performs no automatic safety system function; i.e., it has no scram function. It does provide the operator with a visual indication of neutron level. The consequences of reactivity accidents are functions of the initial neutron flux. The requirement of at least 3 counts per second assures that any transient, should it occur, begins at or above the initial value of 10^{-8} of rated power used in the analyses of transients in cold conditions. One operable SRM channel would be adequate to monitor the approach to criticality using homogeneous patterns of scattered control rod withdrawal. A minimum of two operable SRM's are provided as an added conservatism.

3. The RBM provides local protection of the core; i.e., the prevention of boiling transition in a local region of the core, for a single rod withdrawal error from a Limiting Control Rod Pattern. The trip point is referenced to power. This power signal is provided by the APRMs. A statistical analysis of many single control rod withdrawal errors has been performed and at the 95/95 level the results show that with the specified trip settings, rod withdrawal is blocked at MCPRs greater than the Safety Limit, thus allowing adequate margin. This analysis assumes a steady state MCPR of 1.20 prior to the postulated rod withdrawal error. The RBM functions are required when core thermal power is greater than 30% and a Limiting Control Rod Pattern exists. When both RBM channels are operating either channel will assure required withdrawal blocks occur even assuming a single failure of one channel. When a Limiting Control Rod Pattern exists, with one RBM channel inoperable for no more than 24 hours, testing of the RBM prior to withdrawal of control rods assures that improper control rod withdrawal will be blocked (Reference 4). Requiring at least half of the normal LPRM inputs to be operable assures that the RBM response will be adequate to protect against rod withdrawal errors, as shown by a statistical failure analysis.

The RBM bypass time delay is set low enough to assure minimum rod movement while upscale trips are bypassed.

A Limiting Control Rod Pattern for rod withdrawal error (RWE) exists when (a) core thermal power is greater than or equal to 30% of rated and less than 90% of rated (30% ≤ P < 90%) and the MCPR is less than 1.70, or (b) core thermal power is greater than or equal to 90% of rated (P ≥ 90%) and the MCPR is less than 1.40.

During the use of such patterns, it is judged that testing of the RBM channel (when one channel is inoperable) prior to withdrawal of such rods to assure its operability will assure that improper withdrawal does not occur.

D. Scram Insertion Times

The control rod system is designed to bring the reactor subcritical at a rate fast enough to prevent fuel damage; i.e., to prevent the MCPR from becoming less than the safety limit.

After initial fuel loading and subsequent refuelings when operating above 950 psig, all control rods shall be scram tested within the constraints imposed by the Technical Specifications and before the 40% power level is reached. The requirements for the various scram time measurements ensure that any indication of systematic problems with rod drives will be investigated on a timely basis.

E. Reactivity Anomalies

During each fuel cycle excess operative reactivity varies as fuel depletes and as any burnable poison in supplementary control is burned. The magnitude of this excess reactivity may be inferred from the critical rod configuration. As fuel burnup progresses, anomalous behavior in the excess reactivity may be detected by comparison of the critical rod pattern at selected base states to the predicted rod inventory at that state. Power operating base conditions provide the most sensitive and directly interpretable data relative to core reactivity. Furthermore, using power operating base conditions permits frequent reactivity comparisons.

Requiring a reactivity comparison at the specified frequency assures that a comparison will be made before the core reactivity change exceeds 1% $\Delta k/k$. Deviations in core reactivity greater than 1% $\Delta k/k$ are not expected and require thorough evaluation. One percent reactivity limit is considered safe since an insertion of the reactivity into the core would not lead to transients exceeding design conditions of the reactor system.

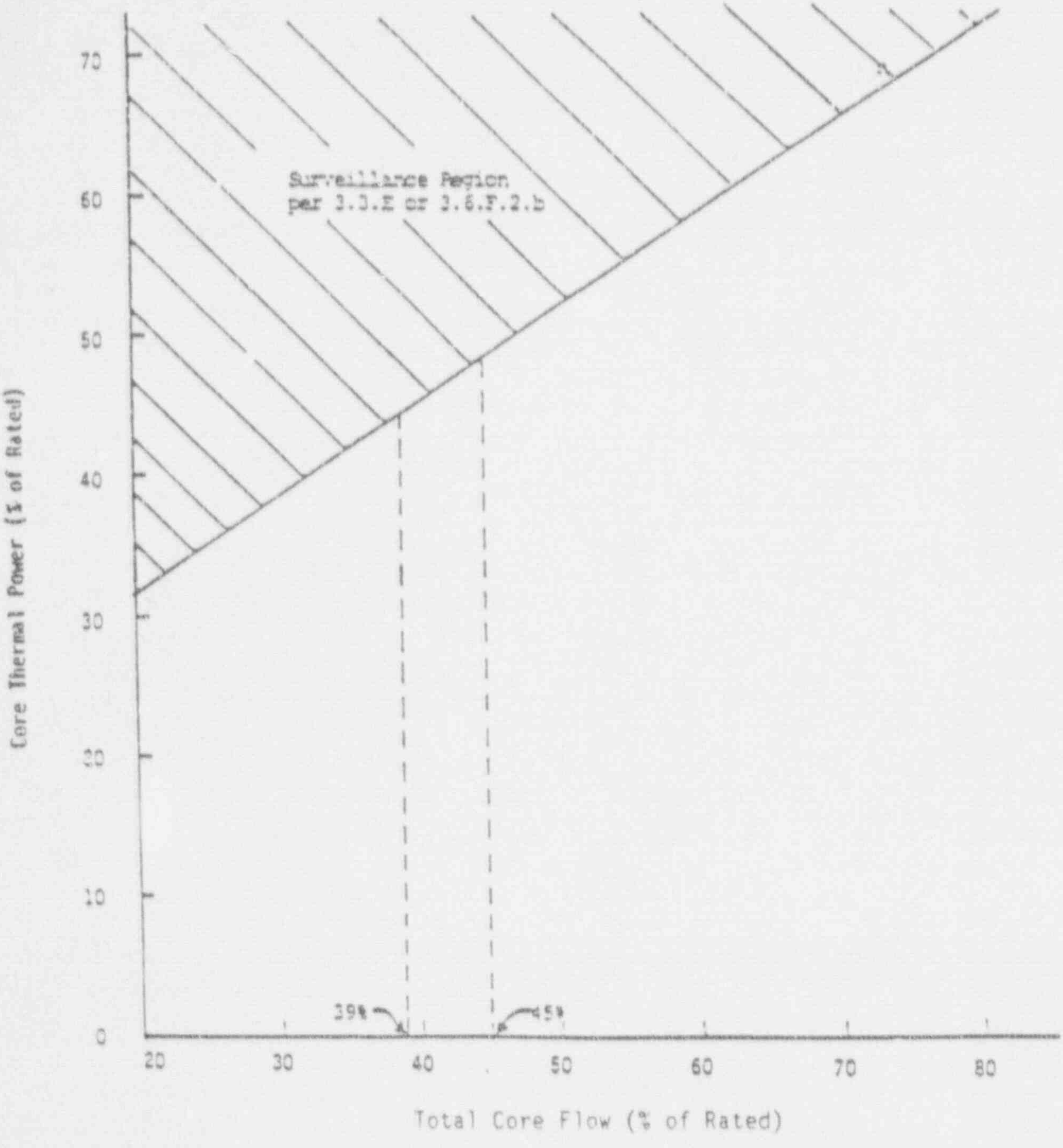
F. Recirculation Pumps

APRM and/or LPRM oscillations in excess of those specified in section 3.3.E could be an indication that a condition of thermal hydraulic instability exists and that appropriate remedial action should be taken. These specifications are based upon the guidance of GE SIL #380, Rev. 1, 2/10/84.

3.3 and 4.3 REFERENCES

1. Banked Position Withdrawal Sequence, NEDO-21231, January 1977.
2. General Electric Standard Application for Reactor Fuel, NEDE-24011-P-A*.
3. General Electric Service Information Letter (SIL) No. 316, Reduced Notch Worth Procedure, November 1979.
4. Average Power Range Monitor, Rod Block Monitor and Technical Specification Improvement (ARTS) Program for the Duane Arnold Energy Center, NEDC-30813-P, December 1984.

*Latest NRC-approved revision.



DAEC Iowa Electric Light and Power Company Technical Specifications
Thermal Power vs Core Flow Limits for Thermal Hydraulic Stability Surveillance Figure 3.3-1

LIMITING CONDITION 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.	3. Observe that any drive which has been uncoupled from and subsequently coupled to its control rod does not go to the overtravel position.
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.	