Proposed Change RTS 117 To

The Duane Arnold Energy Center

Technical Specifications

The holders of License DPR-49 for the Duane Arnold Energy Center propose to amend Appendix A (Technical Specifications) to said license by deleting current pages and replacing them with the attached new proposed pages. A list of the affected pages is attached with the proposed new pages.

Justification for the MCPR changes is contained in Supplemental Reload Licensing Submitted for DAEC Reload 4 (NEDO-24234). The remainder of the changes are justified by the need to make administrative improvements and to adopt more current and generic definitions and terminology.

The MCPR limits proposed for 8 x 8 fuel are higher than those contained in the analyses. As the limits for this type fuel are derived from transient analyses which are expected to change from cycle to cycle, Iowa Electric Light and Power Company has applied a small margin (.02).

Proposed Change RTS 117 to DAEC Technical Specification

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TECHNICAL SPECIFICATIONS

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TECHNICAL SPECIFICATIONS

LIST OF FIGURES

21. Thermal Parameters

a. Minimum Critical 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, within 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.

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING

1.1 FUEL CLADDING INTEGRITY

Applicability:

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

Objective:

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

Specifications:

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

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

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

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

2.1 FUEL CLADDING INTEGRITY

Applicability:

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

Objective:

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

Specifications:

The limiting safety system settings shall be as specified below:

A. Neutron Flux Trips

1. APRM High Flux Scram When In Run Mode.

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

$S \leq (0.66W + 54)$

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

SAFETY LIMIT

16 C. Power Transient

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 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).

LIMITING SAFETY SYSTEM SETTING

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 49×10^8 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. Therefore, 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	$\geq$ 513.5 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 millisecnds of the start of turbine control valve fast closure.

1.1 BASES: FUEL CLADDING INTEGRITY

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

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

The Safety Limit MCPR is generically determined in Reference 1. *

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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 evaluation 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 10^3lbs/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 10^3lbs/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 following 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.

The computer provided with Duane Arnold has a sequence annunciation program which will indicate the sequence in which events such as scram, APRM trip initiation, pressure scram initiation, etc., occur. This program also indicates when the scram setpoint is cleared. This will provide information on how long a scram condition exists and thus provide some measure of the energy added during a transient. Thus, computer information normally will be available for analyzing scrams; however, if the computer information should not be available for any scram analysis, Specification 1.1.C will be relied on to determine if a Safety Limit has been violated.

D. Reactor Water Level (Shutdown Condition)

During periods when the reactor is shut down, consideration must also be given to water level requirements due to the effect of decay heat. If reactor water level should drop below the top of the active fuel during this time, the ability to cool the core is reduced. This reduction in core cooling capability could lead to elevated cladding temperatures and clad perforation. The core can be cooled sufficiently should the water level be reduced to two-thirds the core height. Establishment of the safety limit at 12 inches above the top of the fuel* provides adequate margin. This level will be continuously monitored.

*Top of the active fuel zone is defined to be 344.5 inches above vessel zero (See Bases 3.2).

1.1 REFERENCES

1. "Generic Reload Fuel Application," NEDE-24011-P-A and NEDO-24011-A.*

*Approved Revision at time reload analyses are performed.

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

The abnormal operational transients applicable to operation of the Duane Arnold Energy Center have been analyzed throughout the spectrum of planned operating conditions up to the thermal power condition of 1658 MWt. The analyses were based upon plant operation in accordance with the operating map given in Figure 3.7-1 of the FSAR. In addition, 1658 MWt is the licensed maximum power level of the Duane Arnold Energy Center, and this represents the maximum steady state power which shall not knowingly be exceeded.

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

The absolute value of the void reactivity coefficient used in the analysis is conservatively estimated to be about 25% greater than the nominal maximum absolute value expected to occur during the core lifetime. The scram worth used has been derated to be equivalent to approximately 80% of the total scram worth of the control rods. The scram delay time and rate of rod insertion allowed by the analyses are conservatively set equal to the longest delay and slowest insertion rate acceptable by Technical Specifications. The effect of scram worth, scram delay time and rod insertion rate, all conservatively applied, are of greatest significance in the early portion of the negative reactivity insertion. The rapid insertion of negative reactivity is assured by the time requirements for 25.6% and 46.4% insertion. By the time the rods are 67.2% inserted, approximately four dollars of negative reactivity have been inserted which strongly turns the transient, and accomplishes the desired effect. The times for 4.7% and 88.1% insertion are given to assure proper completion of the expected performance in the earlier portion of the transient, and to establish the ultimate fully shutdown steady-state condition.

For analyses of the thermal consequences of the transients the MCPRs stated in section 3.12 as a limiting condition of operation bound those which are conservatively assumed to exist prior to initiation of the transients.

This choice of using conservative values of controlling parameters and initiating transients at the design power level produces more conservative results than would be obtained by using expected values of control parameters and analyzing at higher power levels.

Steady-state operation without forced recirculation will not be permitted, except during special testing. The analysis to support operation at various power and flow relationships has considered operation with either one or two recirculation pumps.

In summary:

i. The abnormal operational transients have been analyzed to a power level of 1658 MWt.

ii. The licensed maximum power level is 1658 MWt.

iii. Analyses of transients employ adequately conservative values of the controlling reactor parameters.

during operation. Reducing this operating margin would increase the frequency of spurious scrams which have an adverse effect on reactor safety because of the resulting thermal stresses. Thus, the APRM scram trip setting was selected because it provides adequate margin for the fuel cladding integrity Safety Limit yet allows operating margin that reduces the possibility of unnecessary scrams.

The scram trip setting must be adjusted to ensure that the LHGR transient peak is not increased for any combination of MFLPD and reactor core thermal power. The scram setting is adjusted in accordance with the formula in Specification 2.1.A.1, when the maximum fraction of limiting power density is greater than the fraction of rated power. This adjustment may be accomplished by increasing the APRM gain and thus reducing the slope and intercept point of the flow referenced APRM High Flux Scram Curve by the reciprocal of the APRM gain change.

Analyses of the limiting transients show that no scram adjustment is required to assure MCPR greater than or equal to safety limit when the transient is initiated from MCPR $\geq$ values as indicated in Table 3.12.2.

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 as the flow decreases for the specified trip setting versus flow relationship; therefore the worst case MCPR which could occur during steady-state operation is at 108% of rated thermal power because of the APRM rod block trip setting. The actual power distribution in the core is established by specified control rod sequences and is monitored continuously by the in-core LPRM system. As with the APRM scram trip setting, the APRM rod block trip setting is adjusted downward if the maximum fraction of limiting power density exceeds the fraction of rated power, thus preserving the APRM rod block safety margin. As with the scram setting, this may be accomplished by adjusting the APRM gain.

4. 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 an IRM scram would result in a reactor shutdown well before any Safety Limit is exceeded.

that occur during normal or inadvertent isolation valve closure. With the scrams set at 10 percent of valve closure, neutron flux does not increase. To protect the main condenser against overpressure, a loss of condenser vacuum initiates automatic closure of the main steam isolation valves.

G. H. and I. <u>Reactor Low Water Level Setpoint for Initiation of HPCI and RCIC, Closing Main Steam Isolation Valves, and Starting LPCI and Core Spray Pumps</u>

These systems maintain adequate coolant inventory and provide core cooling with the objective of preventing excessive clad temperatures. The design of these systems to adequately perform the intended function is based on the specified low level scram set-
16 point and initiation setpoints. Transient analyses demonstrate that these conditions result in adequate safety margins for both the fuel and the system pressure.

2.1 REFERENCES

1. "Generic Reload Fuel Application," NEDE-24011-P-A* or NEDO-24011-A.

*Approved revision number at time analyses are performed.

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LIMITING CONDITION FOR OPERATION

3.1 REACTOR PROTECTION SYSTEM

Applicability:

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

Objective:

To assure the operability of the reactor protection system.

Specification:

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

SURVEILLANCE REQUIREMENTS

4.1 REACTOR PROTECTION SYSTEM

Applicability:

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

Objective:

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

Specification:

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

2. Response time measurements (from actuation of sensor contacts or trip point to de-energization of scram solenoid relay) are not part of the normal instrument calibration. The measurement of response time will be performed once per operating cycle.

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

TABLE 3.1-1

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENT

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

TABLE 3.1-1 (Continued)

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENT

Minimum No. of Operable Instrument Channels for Trip System (1)	Trip Function	Trip Level Setting	Modes in Which Function Must be Operable: Refuel (6)	Startup	Run	Number of Instrument Channels Provided By Design	Action (1)
2	High Drywell Pressure	$\leq$2.0 psig	X(7)	X(8)	X	4 Instrument Channels	A
2	Reactor Low Water Level	$\geq$+12" Indicated Level	X	X	X	4 Instrument Channels	A
2	High Water Level in Scram Discharge Volume	$\leq$ 60 Gallons	X(2)	X	X	4 Instrument Channels	A
2	Main Steam Line High Radiation	$\leq$3 x Normal Rated Power Background	X	X	X	4 Instrument Channels	A
4	Main Steam Line Isolation Valve Closure	$\leq$ 10% Valve Closure	X (3)(13)	X (3)(13)	X (13)	8 Instrument Channels	A or C
2	Turbine Control Valve Fast Closure (Loss of Control Oil Pressure)	Within 30 milliseconds of the start of Control Valve Fast Closure			X(4)	4 Instrument Channels	A or D
4	Turbine Stop Valve Closure	$\leq$10% Valve Closure			X(4)	8 Instrument Channels	A or D
2	First Stage Pressure Permissive	Bypass below 192 psig at shell	X	X	X	4 Instrument Channels	A or O

*Alarm setting $\leq$1.5 X Normal Rated Power Background

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

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

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

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

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

12. See Subsection 2.1.A.1.

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

14. Deleted.

NOTES FOR TABLE 4.1-1

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

2. A description of the three groups is included in the Bases of this Specification.

3. Functional tests are not required on the part of the system that is not required to be operable or are tripped.

   If tests are missed on parts not required to be operable or are tripped, then they shall be performed prior to returning the system to an operable status.

4. This instrumentation is exempted from the instrument channel test definition. This instrument channel functional test will consist of injecting a simulated electrical signal into the measurement channels.

Experience with passive type instruments in generating stations and substations indicates that the specified calibrations are adequate. For those devices which employ amplifiers, etc., drift specifications call for drift to be less than 0.4%/month; i.e., in the period of a month a maximum drift of 0.4% could occur, thus providing for adequate margin.

For the APRM system, drift of electronic apparatus is not the only consideration in determining a calibration frequency. Change in power distribution and loss of chamber sensitivity dictate a calibration every seven days. Calibration on this frequency assures plant operation at or below thermal limits.

A comparison of Tables 4.1-1 and 4.1-2 indicates that two instrument channels have not been included in the latter table. These are: mode switch in shutdown and manual scram. All of the devices or sensors associated with these scram functions are simple on-off switches and, hence, calibration during operation is not applicable.

2. The peak heat flux is checked once per day to determine if the APRM scram requires adjustment. This will normally be done by checking the LPRM readings. Only a small number of control rods are moved daily and thus the power distribution is

NOTES FOR TABLE 3.2-A

1. Whenever Primary Containment integrity is required by Subsection 3.7, there shall be two operable or tripped trip systems for each function.

2. If the first column cannot be met for one of the trip systems, that trip system shall be tripped or the appropriate action listed below shall be taken.

   A. Initiate an orderly shutdown and have the reactor in Cold Shutdown Condition in 24 hours.

   B. Initiate an orderly load reduction and have Main Steam Lines isolated within eight hours.

   C. Isolate Shutdown Cooling.

   D. Isolate Reactor Water Cleanup System.

3. Instrument setpoint corresponds to 170" above top of active fuel.*

4. Instrument setpoint corresponds to 119.5" above top of active fuel.*

*Top of the active fuel zone is defined to be 344.5 inches above vessel zero (see Bases 3.2).

5. Two required for each steam line.

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

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

8. Deleted

NOTES FOR TABLE 3.2-B

1. Whenever any CSCS subsystem is required by Subsection 3.5 to be operable, there shall be two operable trip systems. If the first column cannot be met for one of the trip systems, that trip system shall be placed in the tripped condition or the reactor shall be placed in the Cold Shutdown Condition within 24 hours.

2. Close isolation valves in RCIC subsystem.

3. Close isolation valves in HPCI subsystem.

4. Instrument setpoint corresponds to 18.5" above the top of active fuel.*

5. HPCI has only one trip system for these sensors.

6. The relay drop-out voltage will be measured once per operating cycle and the data examined for evidence of relay deterioration.

*Top of the active fuel zone is defined to be 344.5 inches above vessel zero (see Bases 3.2).

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)	$\leq(0.66W + 42)\left(\frac{FRP}{MFLPD}\right)$ (2)	6 Inst. Channels	(1)
2	APRM Upscale (Not in Run Mode)	$\leq$12 indicated on scale	6 Inst. Channels	(1)
2	APRM Downscale	$\geq$5 indicated on scale	6 Inst. Channels	(1)
1 (7)	Rod Block Monitor (Flow Biased)	$\leq(0.66W + 39)$	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.

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

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

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

Many of the reactor water level trip settings are defined or described in terms of "inches above the top of the actve fuel." Because the reload fuel is longer than the initial core, the core is now composed of fuel bundles of differing lengths and the term "top of the active fuel" no longer has a definite meaning. Since the basis of all safety analyses is the absolute level (inches above vessel zero) of the trip settings, the "top of the active fuel" has been defined to be 344.5 inches above vessel zero. This definition is the same as that given by the FSAR for the initial core and maintains the consistency between the various level definitions given in the FSAR and the technical specifications.

The low water level instrumentation set to trip at 170" above the top of the active fuel closes all isolation valves except those in Groups 1, 6, 7 and 9 (see notes to Table 3.7-3 for isolation valve groups). Details of valve grouping and required closing times are given in Specification 3.7. For valves which isolate at this level this trip setting is at reduced flow. The APRM provides gross core protection; i.e., limits the gross core power increase from withdrawal of control rods in the normal withdrawal sequence. The trips are set so that MCPR is maintained greater than safety limit.

The RBM rod block function 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 IRM rod block function provides local as well as gross core protection. The scaling arrangement is such that trip setting is less than a factor of 10 above the indicated level.

A downscale indication on an APRM or IRM is an indication the instrument has failed or the instrument is not sensitive enough. In either case the instrument will not respond to changes in control rod motion and thus, control rod motion is prevented. The downscale trips are set at 5 indicated on scale for APRM's and 5/125 full scale for IRM's.

The flow comparator and scram discharge volume high level components have only one logic channel and are not required for safety. The flow comparator must be bypassed when operating with one recirculation water pump.

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_∞. Therefore, an additional margin is included in the shutdown margin test to account for the fact that the "analytically strongest" rod is

3.3 and 4.3 REFERENCES

1) NEDO 24087-3, 78NED265, Class 1, June 1978 "General Electric Boiling Water Reactor Reload 3 (Cycle 4) Licensing Amendment for Duane Arnold Energy Center, Supplement 3: Application of Measured Scram Times".

3.4 BASES

Standby Liquid Control System

1. The purpose of the liquid control system is to provide the capability of bringing the reactor from full power to a cold, xenon-free shutdown condition assuming that none of the withdrawn control rods can be inserted. To meet this objective, the liquid control system is designed to inject a quantity of boron that produces a concentration of 600 ppm of boron in the reactor core in less than 96 minutes. The 600 ppm concentration in the reactor core is required to bring the reactor from full power to a subcritical condition, considering the hot to cold reactivity difference, xenon poisoning, analytical biases and uncertainties, etc. The time requirement for inserting the boron solution was selected to override the rate of reactivity insertion caused by cooldown of the reactor following the xenon poison peak.

   The minimum limitation on the relief valve setting is intended to prevent the recycling of liquid control solution via the lifting of a relief valve at too low a pressure. The upper limit on the relief valve settings provides system protection from overpressure.

the direct scram (valve position scram) results in a peak vessel pressure less than the code allowable overpressure limit of 1375 psig if a flux scram is assumed.

The analyses of the plant isolation transients are evaluated in each reload analyses. These analyses show that the six relief valves assure margin below the setting of the safety valves such that the safety valves would not be expected to open during any anticipated normal transient. These analyses verify that peak system pressure is limited to greater than a 125 psi margin to the allowed vessel overpressure of 1375 psig.

Experience in relief and safety valve operation shows that a testing of 50 percent of the valves per year is adequate to

LIMITING CONDITION FOR OPERATION

3.12 CORE THERMAL LIMITS

Applicability

The Limiting Conditions for Operation associated with the fuel rods apply to those parameters which monitor the fuel rod operating conditions.

Objective

The Objective of the Limiting Conditions for Operation is to assure the performance of the fuel rods.

Specifications

A. Maximum Average Planar Linear Heat Generation Rate (MAPLHGR)

During reactor power operation, the actual MAPLHGR for each type of fuel as a function of average planar exposure shall not exceed the limiting value shown in Figs. 3.12-2, -3, -4, -5, -6, and 7. If at any time during reactor power operation it is determined by normal surveillance that the limiting value for MAPLHGR (LAPLHGR) is being exceeded, action shall then be initiated within 15 minutes to restore operation to within the prescribed limits. If the MAPLHGR (LAPLHGR) is not returned to within the prescribed limits within two hours, the reactor shall be brought to the cold shutdown condition within 36 hours. Surveillance and corresponding action shall continue until the prescribed limits are again being met.

SURVEILLANCE REQUIREMENT

4.12 CORE THERMAL LIMITS

Applicability

The Surveillance Requirements apply to the parameters which monitor the fuel rod operating conditions.

Objective

The Objective of the Surveillance Requirements is to specify the type and frequency of surveillance to be applied to the fuel rods.

Specifications

A. Maximum Average Planar Linear Heat Generation Rate (MAPLHGR)

The MAPLHGR for each type of fuel as a function of average planar exposure shall be determined daily during reactor operation at $\geq$ 25% rated thermal power.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
B. Linear Heat Generation Rate (LHGR)	B. Linear Heat Generation Rate (LHGR)
1. During reactor power operation, the linear heat generation rate (LHGR) of any rod in any 7x7 fuel assembly at any axial location shall not exceed the maximum allowable LHGR as calculated by the following equation:	The LHGR as a function of core height shall be checked daily during reactor operation at $\geq$ 25% rated thermal power.

$$LHGR_{max} \leq LHGR_d[1 - \{(\Delta P/P)_{max}(L/LT)\}]$$

$LHGR_d$ = Design LHGR = 18.5 KW/ft (7x7 array)

$(\Delta P/P_{max}$ = Maximum power spiking penalty = 0.026

LT = Total core length - 12 feet

L = Axial position above bottom of core.

2. During reactor power operation the linear heat generation rate (LHGR) of any rod in any 8x8 fuel assembly shall not exceed 13.4 KW/ft.

If at any time during reactor power operation it is determined by normal surveillance that the limiting value for LHGR is being exceeded, action shall then be initiated within 15 minutes to restore operation to within the prescribed limits. If the LHGR is not returned to within the prescribed limits within two hours, the reactor shall be brought to the cold shutdown condition within 36 hours. Surveillance and corresponding action shall continue until the prescribed limits are again being met.

3.12 BASES: CORE THERMAL LIMITS

A. Maximum Average Planar Linear Heat Generation Rate (MAPLHGR)

This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in 10 CFR Part 50, Appendix K.

The peak cladding temperature following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all rods of a fuel assembly at any axial location and is only dependent secondarily on the rod to rod power distribution within an assembly. Since expected local variations in power distribution within a fuel assembly affect the calculated peak clad temperature by less than ± 20°F relative to the peak temperature for a typical fuel design, the limit on the average linear heat generation rate is sufficient to assure that calculated temperatures are within the 10 CFR Part 50, Appendix K limit.

The calculational procedure used to establish the MAPLHGRs is based on a loss-of-coolant accident analysis. The analysis was performed using General Electric (GE) calculational models which are consistent with the requirements of Appendix K to 10 CFR Part 50.

B. Linear Heat Generation Rate (LHGR)

This specification assures that the linear heat generation rate in any rod is less than the design linear heat generation rate and that the fuel cladding 1% plastic diametral strain linear heat generation rate is not exceeded during any abnormal operating transient if fuel pellet densification is postulated. The power spike penalty specified is based on the analysis presented in Section 3.2.1 of Reference 3 and in References 4 and 5, and assumes a linearly increasing variation in axial gaps between core bottom and top, and assures with a 95% confidence, that no more than one fuel rod exceeds the design linear heat generation rate due to power spiking. The LHGR as a function of core height shall be checked daily during reactor operation at $\geq$ 25% power to determine if fuel burnup, or control rod movement has caused changes in power distribution. For LHGR to be a limiting value below 25% rated thermal power, the MTPF would have to be greater than 10 which is precluded by a considerable margin when employing any permissible control rod pattern.

C. Minimum Critical Power Ratio (MCPR)

1. Operating Limit MCPR

The required operating limit MCPR's at steady state operating conditions as specified in Specification 3.12.C are derived from the established fuel cladding integrity Safety Limit MCPR value, and an analysis of abnormal operational transients[1]. For any abnormal operating transient analysis evaluation with the initial condition of the reactor being at the steady state operating limit it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient assuming instrument trip settings given in Specification 2.1.

To assure that the fuel cladding integrity Safety Limit is not exceeded during any anticipated abnormal operational transient, the most limiting transients have been analyzed to determine which result in the largest reduction in critical power ratio (CPR). The type of transients evaluated were loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease.

The limiting transient, which determines the required steady state MCPR limit, is the transient which yields the largest ΔCPR. The minimum operating limit MCPR of Specification 3.12.C bounds the sum of the safety limit MCPR and the largest ΔCPR.

DELETED

TABLE 3.12-2

MCPR LIMITS

Fuel Type	
7 x 7	1.25
8 x 8	1.24
8 x 8R	1.26

3.12 REFERENCES

1. Duane Arnold Energy Center Loss-of-Coolant Accident Analysis Report, NEDO-21082-02-1A, Class I, July 1977, Appendix A.

2. "Generic Reload Fuel Application," NEDE-24011-P-A**.

3. "Fuel Densification Effects on General Electric Boiling Water Reactor Fuel," Supplements 6, 7 and 8, NEDM-19735, August 1973.

4. Supplement 1 to Technical Reports on Densifications of General Electric Reactor Fuels, December 14, 1973 (AEC Regulatory Staff).

5. Communication: V. A. Moore to I. S. Mitchell, "Modified GE Model for Fuel Densification," Docket 50-321, March 27, 1974.

6. R. B. Linford, Analytical Methods of Plant Transient Evaluations for the GE BWR, February 1973 (NEDO-10802).

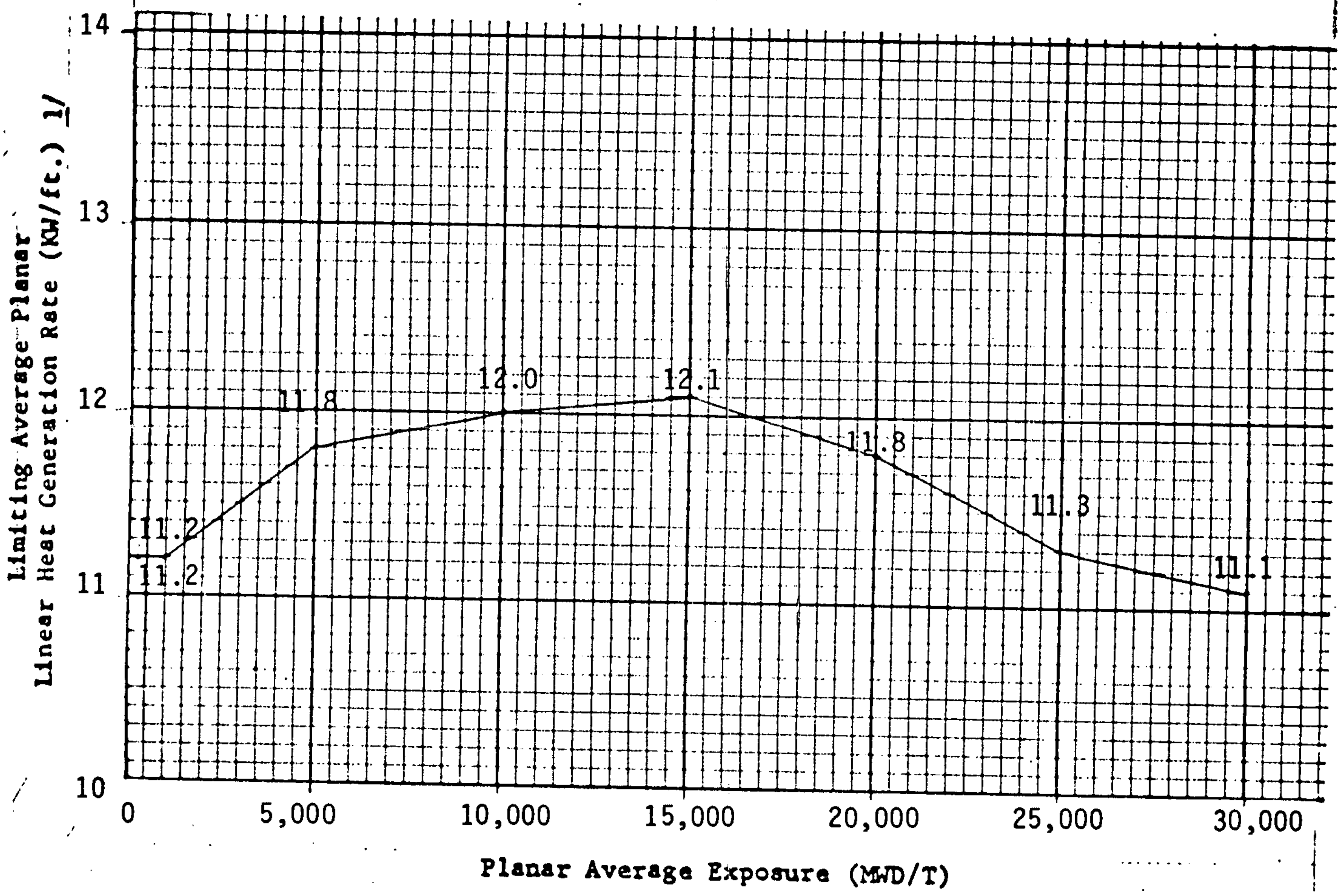
7. General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR Part 50, Appendix K, NEDE-20566 (Draft), August 1974.

8. Boiling Water Reactor Reload-3 Licensing Amendment for Duane Arnold Energy Center, NEDO-24087, 77 NED 359, Class 1, December 1977.

9. Boiling Water Reactor Reload-3 Licensing Amendment for Duane Arnold Energy Center, Supplement 2: Revised Fuel Loading Accident Analysis, NEDO-24087-2.

10. Boiling Water Reactor Reload-3 Licensing Amendment for Duane Arnold Energy Center, Supplement 5: Revised Operating Limits for Loss of Feedwater Heating, NEDO-24987-5.

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



1/ When core flow is equal to or less than 70% of rated, the MAPLHGR shall not exceed 95% of the limiting values shown.

DUANE ARNOLD ENERGY CENTER

IOWA ELECTRIC LIGHT AND POWER COMPANY

TECHNICAL SPECIFICATIONS

LIMITING AVERAGE PLANAR LINEAR HEAT GENERATION RATE AS A FUNCTION OF PLANAR AVERAGE EXPOSURE

FUEL TYPE: P8DP289

FIGURE 3.12-7