DEC 5 1980

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G. Lainas	J. Heltemes, AEOD
J. Roe	NSIC
S. Norris	TERA
R. Bevan	Chairman, ASLAB
OELD	

Docket No. 50-254

Mr. J. S. Abel
Director of Nuclear Licensing
Commonwealth Edison Company
P. O. Box 767
Chicago, Illinois 60690

Dear Mr. Abel:

In response to your application dated September 2, 1980, supplemented by letter dated October 3, 1980, the Commission has issued the enclosed Amendment No. 61 to Facility Operating License No. DPR-29 for the Quad Cities Nuclear Power Station, Unit 1.

This amendment (1) authorizes changes to the plant Technical Specifications which you proposed in your September 2, 1980 submittal, and (2) makes a minor editorial change to the plant Technical Specifications.

Copies of the Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

Original Signed by
T. A. Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #2
Division of Licensing

Enclosures:
1. Amendment No. 61 to DPR-29
2. Safety Evaluation
3. Notice

cc w/enclosures:
See next page

8012300786

P

*SEE PREVIOUS YELLOW FOR CONCURRENCE

OFFICE	ORB #2	ORB #2	AD:OR	OELD	ORB #2	
SURNAME	*SNorris	*Bevan/*Alexion	*TNovak	*RGoddard		
DATE	12/2/80	12/2/80	12/3/80	12/4/80	12/5/80	

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Docket No. 50-254

Mr. J. S. Abel
Director of Nuclear Licensing
Commonwealth Edison Company
P. O. Box 767
Chicago, I-linois 60690

Dear Mr. Abel:

In response to your application dated September 2, 1980, supplemented by letter dated October 3, 1980, the Commission has issued the enclosed Amendment No. to Facility Operating License No. DPR-29 for the Quad Cities Nuclear Power Station, Unit 1.

This amendment (1) authorizes changes to the plant Technical Specifications which you proposed to support your review of future reloads for Quad Cities, Unit 1, under provisions of 10 CFR 50.59 and (2) makes a minor editorial change to the plant Technical Specifications.

Copies of the Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

Thomas A. Ippolito, Chief
Operating Reactors Branch #2
Division of Licensing

Enclosures:
1. Amendment No. to DPR-29
2. Safety Evaluation
3. Notice

cc w/enclosures:
See next page

FR Notice & Amdt: concur w/ change per conversation w/ Scinto 12/4

OFFICE	ORB #2	ORB #2	AD:OR	OELD	ORB #2	
SURNAME	SNorris:kf	RBevan	TNovak		TIppolito	
DATE	12/2/80	12/2/80	12/3/80	12/4/80	12/ /80	

OPERATING REACTORS BRANCH #2
AMENDMENT ROUTE SLIP

1. ~~S. Norris - concurrence~~
2. ~~T. ALEXION - PM concurrence~~
3. ~~T. Ippolito - concurrence~~
4. ~~T. Novak - concurrence (OELD review is/is not requested for SE)~~
5. R. Purple - concurrence
6. ~~OELD - concurrence~~
7. ~~T. Ippolito - signature~~
8. Karen - dispatch

OK 12/5

Letter to: COMMONWEALTH EDISON CO. transmitting Amendment No(s). 61

Subject: 50.59 RELOAD

1. (a) No correspondence has been received related to the subject of this amendment, or

   b. Correspondence has been received, a copy of which is enclosed, which is or may be related to the subject of this amendment.

2. (a) Approval of this amendment will not result in an irreversible impact, or

   b. Approval of this amendment will result in an irreversible impact. Describe the irreversible impact.

3. (a) Recommend that this amendment be post noticed, or

   b. Recommend that this amendment be prenoticed

Remarks: LICENSEE NEEDS THE PROPOSED AMENDMENTS TO RESTART QUAD CITIES 1 WITH THE CYCLE 6 CORE, PRESENTLY SCHEDULED FOR DEC. 8, 1980.

FROM: T. ALEXION

MAIL STOP 338
EXTENSION 29784

cc:

Mr. D. R. Stichnoth
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Mr. Marcel DeJaegher, Chairman
Rock Island County Board
of Supervisors
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Rock Island, Illinois 61201

Director, Criteria and Standards
Division
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U. S. Environmental Protection Agency
Washington, D. C. 20460

U. S. Environmental Protection Agency
Federal Activities Branch
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ATTN: EIS COORDINATOR
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Chicago, Illinois 60604

Susan N. Sekuler
Assistant Attorney General
Environmental Control Division
188 W. Randolph Street
Suite 2315
Chicago, Illinois 60601

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

COMMONWEALTH EDISON COMPANY

AND

IOWA-ILLINOIS GAS AND ELECTRIC COMPANY

DOCKET NO. 50-254

QUAD CITIES NUCLEAR POWER STATION UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 61
License No. DPR-29

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

   A. The application for amendment by the Commonwealth Edison Company (the Licensee) dated September 2, 1980, as supplemented on October 3, 1980, 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 3.B of Facility License No. DPR-29 is hereby amended to read as follows:

B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 61 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas A. Ippolito, Chief
Operating Reactors Branch #2
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 5, 1980

ATTACHMENT TO LICENSE AMENDMENT NO. 61

FACILITY OPERATING LICENSE NO. DPR-29

DOCKET NO. 50-254

1. Remove the following pages and insert identically numbered pages:

iii	
v	3.3/4.3-3
1.0-2	3.3/4.3-4
1.0-4	3.3/4.3-8
1.1/2.1-1	3.3/4.3-9
1.1/2.1-2	3.3/4.3-10
1.1/2.1-4	3.3/4.3-11
1.1/2.1-5	3.4/4.4-3
1.1/2.1-6	3.5/4.5-7
1.1/2.1-7	3.5/4.5-9
1.1/2.1-8	3.5/4.5-10
1.1/2.1-9	3.5/4.5-11
1.1/2.1-10	3.5/4.5-14
1.1/2.1-11	3.5/4.5-15
1.2/2.2-2	3.5/4.5-18
1.2/2.2-3	
3.1/4.1-1	
3.1/4.1-3	
3.1/4.1-5	
3.1/4.1-7	
3.2/4.2-5	
3.2/4.2-6	
3.2/4.2-7	
3.2/4.2-8	
3.2/4.2-14	
3.2/4.2-15	

2. Page 1.1/2.1-2a is added.

3. Figure 2.1-2 is deleted.

4. Figure 3.5-1 is being replaced by 6 pages.

TABLE OF CONTENTS (Cont'd)

TECHNICAL SPECIFICATIONS

APPENDIX A

LIST OF FIGURES

H. **Limiting Conditions for Operation (LCO)** - The limiting conditions for operation specify the minimum acceptable levels of system performance necessary to assure safe startup and operation of the facility. When these conditions are met, the plant can be operated safely and abnormal situations can be safely controlled.

I. **Limiting Safety System Setting (LSSS)** - The limiting safety system settings are settings on instrumentation which initiate the automatic protective action at a level such that the safety limits will not be exceeded. The region between the safety limit and these settings represents margin, with normal operation lying below these settings. The margin has been established so that with proper operation of the instrumentation, the safety limits will never be exceeded.

K. **Logic System Functional Test** - A logic system functional test means a test of all relays and contacts of a logic circuit from sensor to activated device to ensure all components are operable per design intent. Where possible, action will go to completion; i.e., pumps will be started and valves opened.

L. **Modes of Operation** - A reactor mode switch selects the proper interlocking for the operating or shutdown condition of the plant. Following are the modes and interlocks provided:

1. Shutdown - In this position, a reactor scram is initiated, power to the control rod drives is removed, and the reactor protection trip systems have been deenergized for 10 seconds prior to permissive for manual reset.

2. Refuel - In this position, interlocks are established so that one control rod only may be withdrawn when flux amplifiers are set at the proper sensitivity level and the refueling crane is not over the reactor. Also, the trips from the turbine control valves, turbine stop valves, main steam isolation valves, and condenser vacuum are bypassed. If the refueling crane is over the reactor, all rods must be fully inserted and none can be withdrawn.

3. Startup/Hot Standby - In this position, the reactor protection scram trips, initiated by condenser low vacuum and main steamline isolation valve closure, are bypassed, the low pressure main steamline isolation valve closure trip is bypassed, and the reactor protection system is energized, with IRM and APRM neutron monitoring system trips and control rod withdrawal interlocks in service.

4. Run - In this position the reactor system pressure is at or above 850 psig, and the reactor protection system is energized, with APRM protection and RMB interlocks in service (excluding the 15% high flux scram).

M. **Operable** - A system or component shall be considered operable when it is capable of performing its intended function in its required manner.

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

O. **Operating Cycle** - Interval between the end of one refueling outage for a particular unit and the end of the next subsequent refueling outage for the same unit.

P. **Primary Containment Integrity** - Primary containment integrity means that the drywell and pressure suppression chamber are intact and all of the following conditions are satisfied:

1. All manual containment isolation valves on lines connecting to the reactor coolant system or containment which are not required to be open during accident conditions are closed.

Y. **Shutdown** - The reactor is in a shutdown condition when the reactor mode switch is in the Shutdown position and no core alterations are being performed.

1. Hot Shutdown means conditions as above, with reactor coolant temperature greater than 212° F.
2. Cold Shutdown means conditions as above, with reactor coolant temperature equal to or less than 212° F.

Z. **Simulated Automatic Actuation** - Simulated automatic actuation means applying a simulated signal to the sensor to actuate the circuit in question.

BB. **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.

CC. **Critical Power Ratio (CPR)** - The critical power ratio is the ratio of that assembly power which causes some point in the assembly to experience transition boiling to the assembly power at the reactor condition of interest as calculated by application of the GEXL correlation (reference NEDO-10958).

DD. **Minimum Critical Power Ratio (MCPR)** - The minimum incore critical power ratio corresponding to the most limiting fuel assembly in the core.

EE. **Surveillance Interval** - Each surveillance requirement shall be performed within the specified surveillance interval with:

a. A maximum allowable extension not to exceed 25% of the surveillance interval.
b. A total maximum combined interval time for any 3 consecutive surveillance intervals not to exceed 3.25 times the specified surveillance interval.

FF. 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.

GG. 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).

HH. Fraction of Rated Power (FRP) - The fraction of rated power is the ratio of core thermal power to rated thermal power of 2511 MWth.

1.1/2.1 FUEL CLADDING INTEGRITY

SAFETY LIMIT

Applicability:

The safety limits established to preserve the fuel cladding integrity apply to those variables which monitor the fuel thermal behavior.

Objective:

The objective of the safety limits is to establish limits below which the integrity of the fuel cladding is preserved.

LIMITING SAFETY SYSTEM SETTING

Applicability:

The limiting safety system settings apply to trip settings of the instruments and devices which are provided to prevent the fuel cladding integrity safety limits from being exceeded.

Objective:

The objective of the limiting safety system settings is 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

A. Reactor Pressure > 800 psig and Core Flow > 10% of Rated

The existence of a mimimum 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 ≤ 800 psig)

When the reactor pressure is ≤ 800 psig or core flow is less than 10% of rated, the core thermal power shall not exceed 25% of rated thermal power.

C. Power Transient

1. The neutron flux shall not exceed the scram setting established in Specification 2.1.A for longer than 1.5 seconds as indicated by the process computer.
2. When the process computer is out of service, this safety limit shall be assumed to be exceeded if the neutron flux exceeds the scram setting established by Specification 2.1.A and a control rod scram does not occur.

A. Neutron Flux Trip Settings

The limiting safety system trip settings shall be as specified below:

1. APRM Flux Scram Trip Setting (Run Mode)

When the reactor mode switch is in the Run position, the APRM flux scram setting shall be as shown in Figure 2.1-1 and shall be:

$S \leq (.65W_D + 55)$

with a maximum setpoint of 120% for core flow equal to 98 x 10^6 lb/hr and greater,

where:

S = setting in percent of rated power

W_D = percent of drive flow required to produce a rated core flow of 98 million lb/hr. In the event of operation with a maximum fraction of limiting power density (MFLPD) greater than the fraction of rated power (FRP), the setting shall be modified as follows:

$$S \leq (.65W_D + 55)\left[\frac{FRP}{MFLPD}\right]$$

D. Reactor Water Level (Shutdown Condition)

Whenever the reactor is in the shutdown condition with irradiated fuel in the reactor vessel, the water level shall not be less than that corresponding to 12 inches above the top of the active fuel* when it is seated in the core.

*Top of active fuel is defined to be 360 inches above vessel zero (See Bases 3.2).

Where:

FRP = fraction of rated thermal power (2511 MWt)

MFLPD = maximum fraction of limiting power density where the limiting power density for each bundle is the design linear heat generation rate for that bundle.

The ratio of FRP/MFLPD shall be set equal to 1.0 unless the actual operating value is less than 1.0 in which case the actual operating value will be used.

This adjustment may also be performed by increasing the APRM gain by the inverse ratio, MFLPD/FRP, which accomplishes the same degree of protection as reducing the trip setting by FRP/MFLPD.

2. APRM Flux Scram Trip Setting (Refueling or Startup and Hot Standby Mode)

   When the reactor mode switch is in the Refuel or Startup Hot Standby position, the APRM scram shall be set at less than or equal to 15% of rated neutron flux.

3. IRM Flux Scram Trip Setting

   The IRM flux scram setting shall be set at less than or equal to 120/125 of full scale.

4. When the reactor mode switch is in the startup or run position, the reactor shall not be operated in the natural circulation flow mode.

B. APRM Rod Block Setting

The APRM rod block setting shall be as shown in Figure 2.1-1 and shall be:

$$S \leq (.65W_D + 43)$$

The definitions used above for the APRM scram trip apply. In the event of operation with a maximum fraction limiting power density (MFLPD) greater than the fraction of rated power (FRP), the setting shall be modified as follows:

$$S \leq (.65W_D + 43) \frac{FRP}{MFLPD}$$

The definitions used above for the APRM scram trip apply.

The ratio of FRP to MFLPD shall be set equal to 1.0 unless the actual operating value is less than 1.0, in which case the actual operating value will be used.

This may also be performed by increasing the APRM gain by the inverse ratio, MFLPD/FRP, which accomplishes the same degree of protection as reducing the trip setting by FRP/MFLPD.

C. Reactor low water level scram setting shall be 144 inches above the top of the active fuel* at normal operating conditions.

D. Reactor low water level ECCS initiation shall be 84 inches (+4 inches /-0 inch) above the top of the active fuel* at normal operating conditions.

E. Turbine stop valve scram shall be ≤ 10% valve closure from full open.

F. Turbine control valve fast closure scram shall initiate upon actuation of the fast closure solenoid valves which trip the turbine control valves.

G. Main steamline isolation valve closure scram shall be ≤ 10% valve closure from full open.

H. Main steamline low-pressure initiation of main steamline isolation valve closure shall be ≥ 850 psig.

*Top of active fuel is defined to be 360 inches above vessel zero (See Bases 3.2)

1.1 SAFETY LIMIT BASIS

The fuel cladding integrity limit is set such that no calculated fuel damage would occur as a result of an abnormal operational transient. Because fuel damage is not directly observable, a step-back approach is used to establish a safety limit such that the minimum critical power ratio (MCPR) is no less than the fuel cladding integrity safety limit. MCPR > the fuel cladding integrity safety limit represents a conservative margin relative to the conditions required to maintain fuel cladding integrity.

The fuel cladding is one of the physical barriers which separate radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use-related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions and the protection system safety settings. While fission product migration from cladding perforation is just as measurable as that from use-related cracking, the thermally caused cladding perforations signal a threshold beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding safety limit is defined with margin to the conditions which would produce onset of transition boiling (MCPR of 1.0). These conditions represent a significant departure from the condition intended by design for planned operation. Therefore, the fuel cladding integrity safety limit is established such that no calculated fuel damage shall result from an abnormal operational transient. Basis of the values derived for this safety limit for each fuel type is documented in Reference 1.

A. Reactor Pressure > 800 psig and Core Flow > 10% of Rated

Onset of transition boiling results in a decrease in heat transfer from the cladding and therefore elevated cladding temperature and the possibility of cladding failure. However, the existence of critical power, or boiling transition is not a directly observable parameter in an operating reactor. Therefore, the margin to boiling transition is calculated from plant operating parameters such as core power, core flow, feedwater temperature, and core power distribution. The margin for each fuel assembly is characterized by the critical power ratio (CPR), which is the ratio of the bundle power which would produce onset of transition boiling divided by the actual bundle power. The minimum value of this ratio for any bundle in the core is the minimum critical power ratio (MCPR). It is assumed that the plant operation is controlled to the nominal protective setpoints via the instrumented variables (Figure 2.1-3).

The MCPR fuel cladding integrity safety limit has sufficient conservatism to assure that in the event of an abnormal operational transient initiated from the normal operating condition, more than 99.9% of the fuel rods in the core are expected to avoid boiling transition. The margin between MCPR of 1.0 (onset of transition boiling) and the safety limit, is derived from a detailed statistical analysis considering all of the uncertainties in monitoring the core operating state, including uncertainty in the boiling transition correlation (see e.g., Reference 1). Because the boiling transition correlation is based on a large quantity of full-scale data, there is a very high confidence that operation of a fuel assembly at the condition of MCPR = the fuel cladding integrity safety limit would not produce boiling transition.

However, if boiling transition were to occur, cladding perforation would not be expected. Cladding temperatures would increase to approximately 1100°F, which is below the perforation temperature of the cladding material. This has been verified by tests in the General Electric Test Reactor (GETR), where similar fuel operated above the critical heat flux for a significant period of time (30 minutes) without cladding perforation.

If reactor pressure should ever exceed 1400 psia during normal power operation (the limit of applicability of the boiling transition correlation), it would be assumed that the fuel cladding integrity safety limit has been violated.

In addition to the boiling transition limit (MCPR) operation is constrained to a maximum LHGR = 17.5 kw/ft for 7 x 7 fuel and 13.4kw/ft for all 8x8 fuel types. This constraint is established by Specification 3.5.J. to provide adequate safety margin to 1% plastic strain for abnormal operating transients initiated from high power conditions. Specification 2.1.A.1 provides for equivalent safety margin for transients initiated from lower power conditions by adjusting the APRM flow-biased scram setting by the ratio of FRP/MFLPD.

Specification 3.5.J established the LHGR maximum which cannot be exceeded under steady power operation.

B. Core Thermal Power Limit (Reactor Pressure<800 psia)

At pressures below 800 psia, the core elevation pressure drop (0 power, 0 flow) is greater than 4.56 psi. At low powers and 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 powers and flows will always be greater than 4.56 psi. Analyses show that with a flow of 28 x 10^3 lb/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 lb/hr. 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. At 25% of rated thermal power, the peak powered bundle would have to be operating at 3.86 times the average powered bundle in order to achieve this bundle power. Thus, a core thermal power limit of 25% for reactor pressures below 800 psia is conservative.

C. Power Transient

During transient operation the heat flux (thermal power-to-water) would lag behind the neutron flux due to the inherent heat transfer time constant of the fuel, which is 8 to 9 seconds. Also, the limiting safety system scram settings are at values which will not allow the reactor to be operated above the safety limit during normal operation or during other plant operating situations which have been analyzed in detail. In addition, control rod scrams are such that for normal operating transients, the neutron flux transient is terminated before a significant increase in surface heat flux occurs. Control rod scram times are checked as required by Specification 4.3.C.

Exceeding a neutron flux scram setting and a failure of the control rods to reduce flux to less than the scram setting within 1.5 seconds does not necessarily imply that fuel is damaged; however, for this specification, a safety limit violation will be assumed any time a neutron flux scram setting is exceeded for longer than 1.5 seconds.

If the scram occurs such that the neutron flux dwell time above the limiting safety system setting is less than 1.7 seconds, the safety limit will not be exceeded for normal turbine or generator trips, which are the most severe normal operating transients expected. These analyses show that even if the bypass system fails to operate, the design limit of MCPR = the fuel cladding integrity safety limit is not exceeded. Thus, use of a 1.5 second limit provides additional margin.

The computer provided has a sequence annunciation program which will indicate the sequence in which scrams occur, such as neutron flux, pressure, etc. 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.2 will be relied on to determine if a safety limit has been violated.

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 cladding perforation. The core will be cooled sufficiently to prevent cladding melting 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 whenever the recirculation pumps are not operating.

*Top of the active fuel is defined to be 360 inches above vessel zero (see Bases 3.2).

Amendment No. 61

References

1. "Generic Reload Fuel Applications," NEDE-24011-P-A*

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

2.1 LIMITING SAFETY SYSTEM SETTING BASES

The abnormal operational transients applicable to operation of the units have been analyzed throughout the spectrum of planned operating conditions up to the rated thermal power condition of 2511 MWt. In addition, 2511 MWt is the licensed maximum steady-state power level of the units. This maximum steady-state power level will never 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 model. Conservatism incorporated into the transient analysis is documented in Reference 1. Transient analyses are initiated at the conditions given in this Reference.

The absolute value of the void reactivity coefficient used in the analysis is conservatively estimated to be about 25% greater than the nominal maximum 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 and conservatively set equal to the longest delay and slowest insertion rate acceptable by technical specifications. The effects 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 5% and 20% insertion. By the time the rods are 60% inserted, approximately 4 dollars of negative reactivity have been inserted, which strongly turns the transient and accomplishes the desired effect. The times for 50% and 90% insertion are given to assure proper completion of the expected performance in the earlier portion of the transient, and to establish the ultimate fully shut down steady-state condition.

This choice of using conservative values of controlling parameters and initiating transients at the design power level produces more pessimistic answers than would result 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 startup testing. The analysis to support operation at various power and flow relationships has considered operation with either one or two recirculation pumps.

The bases for individual trip settings are discussed in the following paragraphs.

For analyses of the thermal consequences of the transients, the MCPR's stated in Paragraph 3.5.K as the limiting condition of operation bound those which are conservatively assumed to exist prior to initiation of the transients.

A. Neutron Flux Trip Settings

1. APRM Flux Scram Trip Setting (Run Mode)

The average power range monitoring (APRM) system, which is calibrated using heat balance data taken during steady-state conditions, reads in percent of rated thermal power. Because fission chambers provide the basic input signals, the APRM system responds directly to average neutron flux. During transients, the instantaneous rate of heat transfer from the fuel (reactor thermal power) is less than the instantaneous neutron flux due to the time constant of the fuel. Therefore, during abnormal operational transients, the thermal power of the fuel will be less than that indicated by the neutron flux at the scram setting. Analyses demonstrate that with a 120% scram trip setting, none of the abnormal operational transients analyzed violates the fuel safety limit, and there is a substantial margin from fuel damage. Therefore, the use of flow-referenced scram trip provides even additional margin.

An increase in the APRM scram trip setting would decrease the margin present before the fuel cladding integrity safety limit is reached. The APRM scram trip setting was determined by an analysis of margins required to provide a reasonable range for maneuvering 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 maximum fraction of limiting power density (MFLPD) and reactor core thermal power. The scram setting is adjusted in accordance with the formula in Specification 2.1.A.1, when the MFLPD is greater than the fraction of rated power (FRP). The adjustment may be accomplished by increasing the APRM gain by the reciprocal of FRP/MFLPD. This provides the same degree of protection as reducing the trip setting by FRP/MFLPD by raising the initial APRM readings closer to the trip settings such that a scram would be received at the same point in a transient as if the trip settings had been reduced by $\frac{FRP}{MFLPD}$.

2. APRM Flux Scram Trip Setting (Refuel or Startup/Hot Standby Mode)

For operation in the Startup mode while the reactor is at low pressure, the APRM scram settir of 15% of rated power provides adequate thermal margin between the setpoint and the safety limit, 25% 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. Of all possible source 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 signifi cant percentage of rated power, the rate of power rise is very slow. Generally, the heat flu is in 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% 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% APRM scram remains active until the mode switch is placed in the Run position. This switch occurs when reactor pressure is greater than 850 psig.

3. IRM Flux Scram Trip Setting

The IRM system consists of eight chambers, four in each of the reactor protection system log: channels. The IRM is a 5-decade instrument which covers the range of power level between th: covered by the SRM and the APRM. The 5 decades are broken down into 10 ranges, each being one-half 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 1, the scram setting would be 120 divisions for that range; likewise, 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 set: ting is also ranged up.

The most significant sources of reactivity change during the power increase are due to contr: rod withdrawl. In order to ensure that the IRM provides adequate protection against the single rod withdrawal error, a range of rod withdrawal accidents was analyzed. This analysi: 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.

Additional conservatism was taken in this analysis by assuming that the IRM channel closest to the withdrawn rod is bypassed. The results of this analysis show that the reactor is scramm and peak power limited to 1% of rated power, thus maintaining MCPR above the fuel cladding integrity safety limit. Based on the above analysis, the IRM provides protection against local control rod withdrawal errors and continuous withdrawal of control rods in sequence an provides backup protection for the APRM.

B. APRM Rod Block Trip Setting

Reactor power level may be varied by moving control rods or by varying the recirculation flow rate. The APRM system provides a control rod block to prevent gross rod withdrawal at constant recirculation flow rate to protect against grossly exceeding the MCPR Fuel Cladding Integrity Safety Limit. This rod block trip setting, which is automatically varied with recirculation loop flow rate, prevents an increase in the reactor power level to excessive values due to control rod withdrawal. The flow variable trip setting provides substantial margin from fuel damage, assuming a steady-state operation at the trip setting, over the entire recirculation flow range. The margin to the safety limit increases 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 incore LPRM system. As with 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.

C. Reactor Low Water Level Scram

The reactor low water level scram is set at a point which will assure that the water level used in the bases for the safety limit is maintained. The scram setpoint is based on normal operating temperature and pressure conditions because the level instrumentation is density compensated.

D. Reactor Low Low Water Level ECCS Initiation Trip Point

The emergency core cooling subsystems are designed to provide sufficient cooling to the core to dissipate the energy associated with the loss-of-coolant accident and to limit fuel cladding temperature to well below the cladding melting temperature to assure that core geometry remains intact and to limit any cladding metal-water reaction to less than 1%. To accomplish their intended function, the capacity of each emergency core cooling system component was established based on the reactor low water level scram setpoint. To lower the setpoint of the low water level scram would increase the capacity requirement for each of the ECCS components. Thus, the reactor vessel low water level scram was set low enough to permit margin for operation, yet will not be set lower because of ECCS capacity requirements.

The design of the ECCS components to meet the above criteria was dependent on three previously set parameters: the maximum break size, the low water level scram setpoint, and the ECCS initiation setpoint. To lower the setpoint for initiation of the ECCS could lead to a loss of effective core cooling. To raise the ECCS initiation setpoint would be in a safe direction, but it would reduce the margin established to prevent actuation of the ECCS during normal operation or during normally expected transients.

E. Turbine Stop Valve Scram

The turbine stop valve closure scram trip anticipates the pressure, neutron flux, and heat flux increase that could result from rapid closure of the turbine stop valves. With a scram trip setting of 10% of valve closure from full open, the resultant increase in surface heat flux is limited such that MCPR remains above the MCPR fuel cladding integrity safety limit even during the worst-case transient that assumes the turbine bypass is closed.

F. Turbine Control Valve Fast Closure Scram

The turbine control valve fast closure scram is provided to anticipate the rapid increase in pressure and neutron flux resulting from fast closure of the turbine control valves due to a load rejection and subsequent failure of the bypass, i.e., it prevents MCPR from becoming less than the MCPR fuel cladding integrity safety limit for this transient. For the load rejection without bypass transient from 100% power, the peak heat flux (and therefore LHGR) increases on the order of 15% which provides wide margin to the value corresponding to 1% plastic strain of the cladding.

G. **Reactor Coolant Low Pressure Initiates Main Steam Isolation Valve Closure**

The low-pressure isolation at 850 psig was provided to give protection against fast reactor depressurization and the resulting rapid cooldown of the vessel. Advantage was taken of the scram feature which occurs in the Run mode when the main steamline isolation valves are closed to provide for reactor shutdown so that operation at pressures lower than those specified in the thermal hydraulic safety limit does not occur, although operation at a pressure low than 850 psig would not necessarily constitute an unsafe condition.

H. **Main Steamline Isolation to Valve Closure Scram**

The low-pressure isolation of the main steamlines at 850 psig was provided to give protection against rapid reactor depressurization and the resulting rapid cooldown of the vessel. Advantage was taken of the scram feature in the Run mode which occurs when the main steamline isolation valves are closed to provide for reactor shutdown so that high power operation at low reactor pressures does not occur, thus providing protection for the fuel cladding integrity safety limit. Operation of the reactor at pressures lower than 850 psig requires that the reactor mode switch be in the Startup position, where protection of the fuel cladding integrity safety limit is provided by the IRM and APRM high neutron flux scrams. Thus, the combination of main steamline low-pressure isolation and isolation valve closure scram in the Run mode assures the availability of neutron flux scram protection over the entire range of applicability of the fuel cladding integrity safety limit. In addition, the isolation valve closure scram in the Run mode anticipates the pressure and flux transients which occur during normal or inadvertent isolation valve closure. With the scrams set at 10% valve closure in the Run mode, there is no increase in neutron flux.

I. **Turbine EHC Control Fluid Low-Pressure Scram**

The turbine EHC control system operates using high-pressure oil. There are several points in this oil system where a lost of oil pressure could result in a fast closure of the turbine control valves. This fast closure of the turbine control valves is not protected by the turbine control valve fast closure scram, since failure of the oil system would not result in the fast closure solenoid valves being actuated. For a turbine control valve fast closure, the core would be protected by the APRM and high-reactor pressure scrams. However, to provide the same margins as provided for the generator load rejection on fast closure of the turbine control valves, a scram has been added to the reactor protection system which senses failure of control oil pressure to the turbine control system. This is an anticipatory scram and results in reactor shutdown before any significant increase in neutron flux occurs. The transient response is very similar to that resulting from the turbine control valve fast closure scram. The scram setpoint of 900 psig is set high enough to provide the necessary anticipatory function and low enough to minimize the number of spurious scrams. Normal operating pressure for this system is 1250 psig. Finally, the control valves will not start until the fluid pressure is 600 psig. Therefore, the scram occurs well before valve closure begins.

J. **Condenser Low Vacuum Scram**

Loss of condenser vacuum occurs when the condenser can no longer handle the heat input. Loss of condenser vacuum initiates a closure of the turbine stop valves and turbine bypass valves which eliminates the heat input to the condenser. Closure of the turbine stop and bypass valves causes a pressure transient, neutron flux rise, and an increase in surface heat flux. To prevent the cladding safety limit from being exceeded if this occurs, a reactor scram occurs on turbine stop valve closure in the Run mode. The turbine stop valve closure scram function alone is adequate to prevent the cladding safety limit from being exceeded in the event of a turbine trip transient with bypass closure.

The condenser low vacuum scram is anticipatory to the stop valve closure scram and causes a scram before the stop valves are closed and thus the resulting transient is less severe. Scram occurs in the Run mode at 23-inch Hg vacuum stop valve closure occurs at 20-inch Hg vacuum, and bypass closure at 7-inch Hg vacuum.

References

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

*Approved revision number at time reload analyses are performed

Figure 2.1-2 has been deleted

1.2 SAFETY LIMIT BASES

The reactor coolant system integrity is an important barrier in the prevention of uncontrolled release of fission products. It is essential that the integrity of this system be protected by establishing a pressure limit to be observed for all operating conditions and whenever there is irradiated fuel in the reactor vessel.

The pressure safety limit of 1325 psig as measured by the vessel steam space pressure indicator is equivalent to 1375 psig at the lowest elevation of the reactor coolant system. The 1375 psig value is derived from the design pressures of the reactor pressure vessel and coolant system piping. The respective design pressures are 1250 psig at 575° F and 1175 psig at 560° F. The pressure safety limit was chosen as the lower of the pressure transients permitted by the applicable design codes: ASME Boiler and Pressure Vessel Code Section III for the pressure vessel, and USASI B31.1 Code for the reactor coolant system piping. The ASME Boiler and Pressure Vessel Code permits pressure transients up to 10% over design pressure (110% x 1250 = 1375 psig), and the USASI Code permits pressure transients up to 20% over the design pressure (120% x 1175 = 1410 psig). The safety limit pressure of 1375 psig is referenced to the lowest elevation of the primary coolant system. Evaluation methodology to assure that this safety limit pressure is not exceeded for any reload is documented in Reference 1.

The design basis for the reactor pressure vessel makes evident the substantial margin of protection against failure at the safety pressure limit of 1375 psig. The vessel has been designed for a general membrane stress no greater than 26,700 psi at an internal pressure of 1250 psig; this is a factor of 1.5 below the yield strength of 40,100 psi at 575° F. At the pressure limit of 1375 psig, the general membrane stress will only be 29,400 psi, still safely below the yield strength.

The relationships of stress levels to yield strength are comparable for the primary system piping and provide a similar margin of protection at the established safety pressure limit.

The normal operating pressure of the reactor coolant system is 1000 psig. For the turbine trip or loss of electrical load transients, the turbine trip scram or generator load rejection scram together with the turbine bypass system limits the pressure to approximately 1100 psig (References 2, 3 and 4). In addition, pressure relief valves have been provided to reduce the probability of the safety valves operating in the event that the turbine bypass should fail.

Finally, the safety valves are sized to keep the reactor coolant system pressure below 1375 psig with no credit taken for relief valves during the postulated full closure of all MSIVs without direct (valve position switch) scram. Credit is taken for the neutron flux scram, however.

The indirect flux scram and safety valve actuation, provide adequate margin below the peak allowable vessel pressure of 1375 psig.

Reactor pressure is continuously monitored in the control room during operation on a 1500 psi full-scale pressure recorder.

References

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

2. SAR, Section 11.22

3. Quad Cities 1 Nuclear Power Station first reload license submittal, Section 6.2.4.2, February 1974.

4. GE Topical Report NEDO-20693, General Electric Boiling Water Reactor No. 1 licensing submittal for Quad Cities Nuclear Power Station Unit 2, December 1974.

* Approved revision number at time reload analyses are performed.

2.2 LIMITING SAFETY SYSTEM SETTING BASES

In compliance with Section III of the ASME Code, the safety valves must be set to open at no higher than 103% of design pressure, and they must limit the reactor pressure to no more than 110% of design pressure. Both the high neutron flux scram and safety valve actuation are required to prevent overpressurizing the reactor pressure vessel and thus exceeding the pressure safety limit. The pressure scram is available as backup protection to the high flux scram. Analyses are performed as described in the "Generic Reload Fuel Application," NEDE-24011-P-A (approved revision number at time reload analyses are performed) for each reload to assure that the pressure safety limit is not exceeded. If the high-flux scram were to fail, a high-pressure scram would occur at 1060 psig.

3.1/4.1 REACTOR PROTECTION SYSTEM

LIMITING CONDITIONS FOR OPERATION

Applicability:

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

Objective:

To assure the operability of the reactor protection system.

SURVEILLANCE REQUIREMENTS

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.

SPECIFICATIONS

A. 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 Tables 3.1-1 through 3.1-4. The 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 milliseconds.

B. If, during operation, the maximum fraction of limiting power density exceeds the fraction of rated power when operating above 25% rated thermal power, either:

1. the APRM scram and rod block settings shall be reduced to the values given by the equations in Specifications 2.1.3.1 and 2.1.3. This may also be accomplished by increasing the APRM gain as described therein.

2. the power distribution shall be changed such that the maximum fraction of limiting power density no longer exceeds the fraction of rated power.

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

B. Daily during reactor power operation, the core power distribution shall be checked for maximum fraction of limiting power density (MFLPD) and compared with the fraction of rated power (FRP) when operating above 25% rated thermal power.

C. When it is determined that a channel is failed in the unsafe condition and Column 1 of Tables 3.1-1 through 3.1-3 cannot be met, that trip system must be put in the tripped condition immediately. All other RPS channels that monitor the same variable shall be functionally tested within 8 hours. The trip system with the failed channel may be untripped for a period of time not to exceed 1 hour to conduct this testing. As long as the trip system with the failed channel contains at least one operable channel monitoring that same variable, that trip system may be placed in the untripped position for short periods of time to allow functional testing of all RPS instrument channels as specified by Table 4.1-1. The trip system may be in the untripped position for no more than 8 hours per functional test period for this testing.

gallons. As indicated above, there is sufficient volume in the piping to accommodate the scram without impairment of the scram times or amount of insertion of the control rods. This function shuts the reactor down while sufficient volume remains to accommodate the discharged water and precludes the situation in which a scram would be required but not be able to perform its function adequately.

Loss of condenser vacuum occurs when the condenser can no longer handle heat input. Loss of condenser vacuum initiates a closure of the turbine stop valves and turbine bypass valves, which eliminates the heat input to the condenser. Closure of the turbine stop and bypass valves causes a pressure transient, neutron flux rise, and an increase in surface heat flux. To prevent the cladding safety limit from being exceeded if this occurs, a reactor scram occurs on turbine stop valve closure. The turbine stop valve closure scram function alone is adequate to prevent the cladding safety limit from being exceeded in the event of a turbine trip transient with bypass closure.

The condenser low-vacuum scram is a backup to the stop valve closure scram and causes a scram before the stop valves are closed, thus the resulting transient is less severe. Scram occurs at 23 inches Hg vacuum, stop valve closure occurs at 20 inches Hg vacuum, and bypass closure at 7 inches Hg vacuum.

High radiation levels in the main steamline tunnel above that due to the normal nitrogen and oxygen radioactivity are an indication of leaking fuel. A scram is initiated whenever such radiation level exceeds seven times normal background. The purpose of this scram is to reduce the source of such radiation to the extent necessary to prevent excessive turbine contamination. Discharge of excessive amounts of radioactivity to the site environs is prevented by the air ejector off-gas monitors, which cause an isolation of the main condenser off-gas line provided the limit specified in Specification 3.8 is exceeded.

The main steamline isolation valve closure scram is set to scram when the isolation valves are 10% closed from full open. This scram anticipates the pressure and flux transient which would occur when the valves close. By scramming at this setting, the resultant transient is insignificant.

A reactor mode switch is provided which actuates or bypasses the various scram functions appropriate to the particular plant operating status (reference SAR Section 7.7.1.2). Whenever the reactor mode switch is in the Refuel or Startup/Hot Standby position, the turbine condenser low-vacuum scram and main steamline isolation valve closure scram are bypassed. This bypass has been provided for flexibility during startup and to allow repairs to be made to the turbine condenser. While this bypass is in effect, protection is provided against pressure or flux increases by the high-pressure scram and APRM 15% scram, respectively, which are effective in this mode.

If the reactor were brought to a hot standby condition for repairs to the turbine condenser, the main steamline isolation valves would be closed. No hypothesized single failure or single operator action in this mode of operation can result in an unreviewed radiological release.

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

The IRM system provides protection against excessive power levels and short reactor periods in the startup and intermediate power ranges (reference SAR Sections 7.4.4.2 and 7.4.4.3). A source range monitor (SRM) system is also provided to supply additional neutron level information during startup but has no scram functions (reference SAR Section 7.4.3.2). Thus the IRM is required in the Refuel and Startup/Hot Standby modes. In addition, protection is provided in this range by the APRM 15% scram as discussed in the bases for Specification 2.1. In the power range, the APRM system provides required protection (reference SAR Section 7.4.5.2). Thus, the IRM system is not required in the Run mode, the APRM's cover only the intermediate and power range; the IRM's provide adequate coverage in the startup and intermediate range.

The high-reactor pressure, high-drywell pressure, reactor low water level, and scram discharge volume high level scrams are required for the Startup/Hot Standby and Run modes of plant operation. They are therefore required to be operational for these modes of reactor operation.

The turbine condenser low-vacuum scram is required only during power operation and must be bypassed to start up the unit.

4.1 SURVEILLANCE REQUIREMENTS BASES

A. The minimum functional testing frequency used in this specification is based on a reliability analysis using the concepts developed in Reference 1. This concept was specifically adapted to the one-out-of-two taken twice logic of the reactor protection system. The analysis shows that the sensors are primarily responsible for the reliability of the reactor protection system. This analysis makes use of 'unsafe failure' rate experience at conventional and nuclear power plants in a reliability model for the system. An 'unsafe failure' is defined as one which negates channel operability and which, due to its nature, is revealed only when the channel is functionally tested or attempts to respond to a real signal. Failures such as blown fuses, ruptured bourdon tubes, faulted amplifiers, faulted cables, etc., which result in 'upscale' or 'downscale' readings on the reactor instrumentation are 'safe' and will be easily recognized by the operators during operation because they are revealed by an alarm or a scram.

The channels listed in Tables 4.1-1 and 4.1-2 are divided into three groups respecting functional testing. These are:

1. on-off sensors that provide a scram trip function (Group 1);
2. analog devices coupled with bistable trips that provide a scram function (Group 2); and
3. devices which serve a useful function only during some restricted mode of operation, such as Startup/Hot Standby, Refuel, or Shutdown, or for which the only practical test is one that can be performed at shutdown (Group 3).

The sensors that make up Group 1 are specifically selected from among the whole family of industrial on-off sensors that have earned an excellent reputation for reliable operation. Actual history on this class of sensors operating in nuclear power plants shows four failures in 472 sensor years, or a failure rate of 0.97×10^{-6}/hr. During design, a goal of 0.99999 probability of success (at the 50% confidence level) was adopted to assure that a balanced and adequate design is achieved. The probability of success is primarily a function of the sensor failure rate and the test interval. A 3-month test interval was planned for Group 1 sensors. This is in keeping with good operating practice and satisfies the design goal for the logic configuration utilized in the reactor protection system.

To satisfy the long-term objective of maintaining an adequate level of safety throughout the plant lifetime, a minimum goal of 0.9999 at the 95% confidence level is proposed. With the one-out-of-two taken twice logic, this requires that each sensor have an availability of 0.993 at the 95% confidence level. This level of availability may be maintained by adjusting the test interval as a function of the observed failure history (Reference 1). To facilitate the implementation of this technique, Figure 4.1-1 is provided to indicate an appropriate trend in test interval. The procedure is as follows:

1. Like sensors are pooled into one group for the purpose of data acquisition.
2. The factor M is the exposure hours and is equal to the number of sensors in a group, n, times the elapsed time T ($M = nT$).
3. The accumulated number of unsafe failures is plotted as an ordinate against M as an abscissa on Figure 4.1-1.
4. After a trend is established, the appropriate monthly test interval to satisfy the goal will be the test interval to the left of the plotted points.
5. A test interval of 1 month will be used initially until a trend is established.

Group 2 devices utilize an analog sensor followed by an amplifier and a bistable trip circuit. The sensor and amplifier are active components, and a failure is almost always accompanied by an alarm and an indication of the source of trouble. In the event of failure, repair or substitution can start immediately. An 'as-is' failure is one that 'sticks' midscale and is not capable of going either up or down in response

switches, hence calibration is not applicable; i.e., the switch is either on or off. Based on the above, no calibration is required for these instrument channels.

B. The MFLPD shall be checked once per day to determine if the APRM scram requires adjustment. This may normally be done by checking the LPRM readings, TIP traces, or process computer calculations. Only a small number of control rods are moved daily, thus the peaking factors are not expected to change significantly and a daily check of the MFLPD is adequate.

References

1. I. M. Jacobs, 'Reliability of Engineered Safety Features as a Function of Testing Frequency,' *Nuclear Safety*, Vol. 9, No. 4, pp. 310-312, July-August 1968.

3.2 LIMITING CONDITIONS FOR OPERATION BASES

In addition to reactor protection instrumentation which initiates a reactor scram, protective instrumentation has been provided which initiates action to mitigate the consequences of accidents which are beyond the operator's ability to control, or terminates operator errors before they result in serious consequences. This set of specifications provides the limiting conditions of operation for the primary system isolation function, initiation of the emergency core cooling system, control rod block, and standby gas treatment systems. The objectives of the specifications are (1) to assure the effectiveness of the protective instrumentation when required by preserving its capability to tolerate a single failure of any component of such systems even during periods when portions of such systems are out of service for maintenance, and (2) to prescribe the trip settings required to assure adequate performance. When necessary, one channel may be made inoperable for brief intervals to conduct required functional tests and calibrations. Some of the settings on the instrumentation that initiates or controls core and containment cooling have tolerances explicitly stated where the high and low values are both critical and may have a substantial effect on safety. It should be noted that 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.

Isolation valves are installed in those lines that penetrate the primary containment and must be isolated during a loss-of-coolant accident so that the radiation dose limits are not exceeded during an accident condition. Actuation of these valves is initiated by the protective instrumentation which senses the conditions for which isolation is required (this instrumentation is shown in Table 3.2-1). Such instrumentation must be available whenever primary containment integrity is required. The objective is to isolate the primary containment so that the guidelines of 10 CFR 100 are not exceeded during an accident.

The instrumentation which initiates primary system isolation is connected in a dual bus arrangement. Thus the discussion given in the bases for Specification 3.1 is applicable here.

The low-reactor water level instrumentation is set to trip at >8 inches on the level instrument (top of active fuel is defined to be 360 inches above vessel zero) and after allowing for the full power pressure drop across the steam dryer the low level trip is at 504 inches above vessel zero, or 144 inches above top of active fuel. Retrofit 8x8 fuel has an active fuel length 1.24 inches longer than earlier fuel designs, however, present trip setpoints were used in the LOCA analysis.* This trip initiates closure of Group 2 and 3 primary containment isolation valves but does not trip the recirculation pumps (reference SAR, Section 7.7.2). For a trip setting of 504 inches above vessel zero and a 60-second valve closure time, the valves will be closed before perforation of the cladding occurs even for the maximum break. The setting is, therefore, adequate.

The low-low reactor level instrumentation is set to trip when reactor water level is 444 inches above vessel zero (with top of active fuel defined as 360 inches above vessel zero, -59" is 84 inches above the top of active fuel). This trip initiates closure of Group 1 primary containment isolation valves (reference SAR Section 7.7.2.2) and also activates the ECC subsystems, starts the emergency diesel generator, and trips the recirculation pumps. This trip setting level was chosen to be high enough to prevent spurious operation but low enough to initiate ECCS operation and primary system isolation so that no melting of the fuel cladding will occur and so that postaccident cooling can be accomplished and the guidelines of 10 CFR 100 will not be exceeded. For the complete circumferential break of a 28-inch recirculation line and with the trip setting given above, ECCS initiation and primary system isolation are initiated and in time so meet the above criteria.

The instrumentation also covers the full spectrum of breaks and meets the above criteria.

* Loss of coolant accident analysis for Dresden Unit 2/3 & Quad Cities Units 1/2, NEDO-24146A, April, 1979.

Venturi tubes are provided in the main steamlines as a means of measuring steam flow and also limiting the loss of mass inventory from the vessel during a steamline break accident. In addition to monitoring steam flow, instrumentation is provided which causes a trip of Group 1 isolation valves. The primary function of the instrumentation is to detect a break in the main steamline, thus only Group 1 valves are closed. For the worst-case accident, main steamline break outside the drywell, this trip setting of 120% of rated steam flow, in conjunction with the flow limiters and main steamline valve closure, limits the mass inventory loss such that fuel is not uncovered, fuel temperatures remain less than 1500° F, and release of radioactivity to the environs is well below 10 CFR 100 guidelines (reference SAR Sections 14.2.3.9 and 14.2.3.10).

Temperature-monitoring instrumentation is provided in the main steamline tunnel to detect leaks in this area. Trips are provided on this instrumentation and when exceeded cause closure of Group 1 isolation valves. Its setting of 200° F is low enough to detect leaks of the order of 5 to 10 gpm; thus it is capable of covering the entire spectrum of breaks. For large breaks, it is a backup to high-steam flow instrumentation discussed above, and for small breaks with the resulting small release of radioactivity, gives isolation before the guidelines of 10 CFR 100 are exceeded.

High-radiation monitors in the main steamline tunnel have been provided to detect gross fuel failure. This instrumentation causes closure of Group 1 valves, the only valves required to close for this accident. With the established setting of 7 times normal background and main steamline isolation valve closure, fission product release is limited so that 10 CFR 100 guidelines are not exceeded for this accident (reference SAR Section 12.2.1.7).

Pressure instrumentation is provided which trips when main steamline pressure drops below 850 psig. A trip of this instrumentation results in closure of Group 1 isolation valves. In the Refuel and Startup/Hot Standby modes this trip function is bypassed. This function is provided primarily to provide protection against a pressure regulator malfunction which would cause the control and/or bypass valve to open. With the trip set at 850 psig, inventory loss is limited so that fuel is not uncovered and peak cladding temperatures are much less than 1500° F; thus, there are no fission products available for release other than those in the reactor water (reference SAR Section 11.2.3).

The RCIC and the HPCI high flow and temperature instrumentation are provided to detect a break in their respective piping. Tripping of this instrumentation results in actuation of the RCIC or of HPCI isolation valves. Tripping logic for this function is the same as that for the main steamline isolation valves, thus all sensors are required to be operable or in a tripped condition to meet the single-failure criteria. The trip settings of 200° F and 300% of design flow and valve closure time are such that core uncovery is prevented and fission product release is within limits.

The instrumentation which initiates ECCS action is arranged in a one-out-of-two taken twice logic circuit. Unlike the reactor scram circuits, however, there is one trip system associated with each function rather than the two trip systems in the reactor protection system. The single-failure criteria are met by virtue of the fact that redundant core cooling functions are provided, e.g., sprays and automatic blowdown and high-pressure coolant injection. The specification requires that if a trip system becomes inoperable, the system which it activates is declared inoperable. For example, if the trip system for core spray A becomes inoperable, core spray A is declared inoperable and the out-of-service specifications of Specification 3.5 govern. This specification preserves the effectiveness of the system with respect to the single-failure criteria even during periods when maintenance or testing is being performed.

The control rod block functions are provided to prevent excessive control rod withdrawal so that MCPR does not go below the MCPR Fuel Cladding Integrity Safety Limit.

The trip logic for this function is one out of n; e.g., any trip on one of the six APRM's, eight IRM's four SRM's will result in a rod block. The minimum instrument channel requirements assure sufficient instrumentation to assure that the single-failure criteria are met. The minimum instrument channel requirements for the RBM may be reduced by one for a short period of time to allow for maintenance, testing, or calibration. This time period is only ~3% of the operating time in a month and does not significantly increase the risk of preventing an inadvertent control rod withdrawal.

The APRM rod block function is flow biased and prevents a significant reduction in MCPR, especially during operation at reduced flow. The APRM provides gross core protection, i.e., limits the gross of control rods in the normal withdrawal sequence.

In the refuel and startup/hot standby modes, the APRM rod block function is set at 12% of rated power. This control rod block provides the same type of protection in the Refuel and Startup/Hot Standby modes as the APRM flow-biased rod block does in the Run mode, i.e., prevents control rod withdrawal before a scram is reached.

The RBM rod block function provides local protection of the core, i.e., the prevention of transition boiling in a local region of the core for a single rod withdrawal error from a limiting control rod pattern. The trip point is flow biased. The worst-case single control rod withdrawal error is analyzed for each reload to assure that, with the specific trip settings, rod withdrawal is blocked before the MCPR reaches the fuel cladding integrity safety limit.

Below 30% power, the worst-case withdrawal of a single control rod without rod block action will not violate the fuel cladding integrity safety limit. Thus the RBM rod block function is not required below this power level.

The IRM block function provides local as well as gross core protection. The scaling arrangement is such that the trip setting is less than a factor of 10 above the indicated level. Analysis of the worst-case accident results in rod block action before MCPR approaches the MCPR fuel cladding integrity safety limit.

A downscale indication on an APRM or IRM is an indication the instrument has failed or is not sensitive enough. In either case the instrument will not respond to changes in control rod motion, and the control rod motion is thus prevented. The downscale trips are set at 3/125 of full scale.

The SRM rod block with ≤ 100 CPS and the detector not fully inserted assures that the SRM's are not withdrawn from the core prior to commencing rod withdrawal for startup. The scram discharge volume high water level rod block provides annunciation for operator action. The alarm setpoint has been selected to provide adequate time to allow determination of the cause of level increase and corrective action prior to automatic scram initiation.

For effective emergency core cooling for small pipe breaks, the HPCI system must function, since reactor pressure does not decrease rapidly enough to allow either core spray or LPCI to operate in time. The automatic pressure relief function is provided as a backup to the HPCI in the event the HPCI does not operate. The arrangement of the tripping contacts is such as to provide this function when necessary and minimize spurious operation. The trip settings given in the specification are adequate to assure the above criteria are met (reference SAR Section 6.2.6.3). The specification preserves the effectiveness of the system during periods of maintenance, testing, or calibration and also minimizes the risk of inadvertent operation, i.e., only one instrument channel out of service.

Two air ejector off-gas monitors are provided and, when their trip point is reached, cause an isolation of the air ejector off-gas line. Isolation is initiated when both instruments reach their high trip point or one has an upscale trip and the other a downscale trip. There is a 15-minute delay before the air ejector off-gas isolation valve is closed. This delay is accounted for by the 30-minute holdup time of the off-gas before it is released to the chimney.

Both instruments are required for trip, but the instruments are so designed that any instrument failure gives a downscale trip. The trip settings of the instruments are set so that the chimney release rate limit given in Specification 3.8.A.2 is not exceeded.

Four radiation monitors are provided in the reactor building ventilation ducts which initiate isolation of the reactor building and operation of the standby gas treatment system. The monitors are located in the reactor building ventilation duct. The trip logic is a one-out-of-two for each set, and each set can initiate a trip independent of the other set. Any upscale trip will cause the desired action. Trip settings of 2 mR/hr for monitors in the ventilation duct are based upon initiating normal ventilation isolation and standby gas treatment system operation so that the ventilation stack release rate limit given in Specification 3.8.A.3 is not exceeded. Two radiation monitors are provided on the refueling floor which initiate isolation of the reactor building and operation of the standby gas treatment systems. The trip logic is one-out of two. Trip settings of 100 mR/hr for the monitors on the refueling floor are based upon initiating normal ventilation isolation and standby gas treatment system operation

so that none of the activity released during the refueling accident leaves the reactor building via the normal ventilation stack but that all the activity is processed by the standby gas treatment system.

The instrumentation which is provided to monitor the postaccident condition is listed in Table 3.2-4. The instrumentation listed and the limiting conditions for operation on these systems ensure adequate monitoring of the containment following a loss-of-coolant accident. Information from this instrumentation will provide the operator with a detailed knowledge of the conditions resulting from the accident; based on this information he can make logical decisions regarding postaccident recovery.

The specifications allow for postaccident instrumentation to be out of service for a period of 7 days. This period is based on the fact that several diverse instruments are available for guiding the operator should an accident occur, on the low probability of an instrument being out of service and an accident occurring in the 7-day period, and on engineering judgment.

The normal supply of air for the control room ventilation system comes from outside the service building. In the event of an accident, this source of air may be required to be shut down to prevent high doses of radiation in the control room. Rather than provide this isolation function on a radiation monitor installed in the intake air duct, signals which indicate an accident, i.e., high drywell pressure, low water level, main steamline high flow, or high radiation in the reactor building ventilation duct, will cause isolation of the intake air to the control room. The above trip signals result in immediate isolation of the control room ventilation system and thus minimize any radiation dose.

TABLE 3.2.3

INSTRUMENTATION THAT INITIATES ROD BLOCK

Minimum Number of Operable or Tripped Instrument Channels per Trip System[1]	Instrument	Trip Level Setting
2	APRM upscale (flow bias)[7]	$\leq [0.650W_D + 43^{[2]}] \frac{FRP}{MFLPD}$
2	APRM upscale (Refuel and Startup/Hot Standby mode)	≤12/125 full scale
2	APRM downscale[7]	≥3/125 full scale
1	Rod block monitor upscale (flow bias)[7]	$\leq 0.650W + 42^{[2]}$
1	Rod block monitor downscale[7]	≥3/125 full scale
3	IRM downscale [3] [8]	≥3/125 full scale
3	IRM upscale[8]	≤108/125 full scale
2[5]	SRM detector not in Startup position[4]	≥2 feet below core centerline
3	IRM detector not in Startup position[8]	≥2 feet below core centerline
2[5] [6]	SRM upscale	$\leq 10^5$ counts/sec
2[5]	SRM downscale[9]	$\geq 10^2$ counts/sec
1	High water level in scram discharge volume	≤25 gallons

Notes

1. For the Startup/Hot Standby and Run positions of the reactor mode selector switch, there shall be two operable or tripped trip systems for each function except the SRM rod blocks. IRM upscale and IRM downscale need not be operable in the Run position, APRM downscale, APRM upscale (flow biased), and RBM downscale need not be operable in the Startup/Hot Standby mode. The RBM upscale need not be operable at less than 30% rated thermal power. One channel may be bypassed above 30% rated thermal power provided that a limiting control rod pattern does not exist. For systems with more than one channel per trip system, if the first column cannot be met for one of the two trip systems, this condition may exist for up to 7 days provided that during that time the operable system is functionally tested immediately and daily thereafter; if this condition lasts longer than 7 days the system shall be tripped. If the first column cannot be met for both trip systems, the systems shall be tripped.

2. W_D is the percent of drive flow required to produce a rated core flow of 98 million lb/hr. Trip level setting is in percent of rated power (2511 MWt).

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

4. This function is bypassed when the count rate is ≥100 CPS.

5. One of the four SRM inputs may be bypassed.

6. This SRM function may be bypassed in the higher IRM ranges (ranges 8, 9, and 10) when the IRM upscale rod block is operable

7. Not required to be operable while performing low power physics tests at atmospheric pressure during or after refueling at power levels not to exceed 5 MWt.

8. This IRM function occurs when the reactor mode switch is in the Refuel or Startup/Hot Standby position.

9. This trip is bypassed when the SRM is fully inserted.

TABLE 3.2.4

POSTACCIDENT MONITORING INSTRUMENTATION REQUIREMENTS[2]

Minimum Number of Operable Channels[1] [3]	Parameter	Instrument Readout Location Unit 1	Number Provided	Range
1	Reactor pressure	901-5	1 2	0-1500 psig 0-1200 psig
1	Reactor water level	901-3	2	-100 inches + 200 inches (0 inches is top of fuel) *
1	Torus water temperature	901-21	2	0-200° F
1	Torus air temperature	901-21	2	0-600° F
2[4]	Torus water level, indicator	901-3	1	-25 inches – + 25 inches
	Torus water level, sight glass		1	18 inch range
1	Torus pressure	901-3	1	-5 inches Hg to 5 psig
1	Drywell pressure	901-3	1	-5 inches Hg to 5 psig 0 to 75 psig
2	Drywell temperature	901-21	6	0-600° F
2	Neutron monitoring	901-5	4	0.1-10^6 CPS
2[4]	Torus to drywell differential pressure		2	0-3 psid

Notes

1. Instrument channels required during power operation to monitor postaccident conditions.
2. Provisions are made for local sampling and monitoring of drywell atmosphere.

*Top of active fuel is defined to be 360 inches above vessel zero (See Bases 3.2).

3. The control rod drive housing support system shall be in place during reactor power operation and when the reactor coolant system is pressurized above atmospheric pressure with fuel in the reactor vessel, unless all control rods are fully inserted and Specification 3.3.A.1 is met.

   a. Control rod withdrawal sequences shall be established so that maximum reactivity that could be added by dropout of any increment of any one control blade would be such that the rod drop accident design limit of 280 cal/gm. is not exceeded.

   b. Whenever the reactor is in the Startup/Hot Standby or Run mode below 20% rated thermal power, the rod worth minimizer shall be operable. A second operator or qualified technical person may be used as a substitute for an inoperable rod worth minimizer which fails after withdrawal of at least 12 control rods to the fully withdrawn position. The rod worth minimizer may also be bypassed for low power physics testing to demonstrate the shutdown margin requirements of Specification 3.3.A if a nuclear engineer is present and verifies the step-by-step rod movements of the test procedure.

4. Control rods shall not be withdrawn for startup or refueling unless at least two source range channels have an observed count rate equal to or greater than three counts per second and these SRM's are fully inserted.

5. During operation with limiting control rod patterns, as determined by the nuclear engineer, either:

   a. both RBM channels shall be operable,

   b. control rod withdrawal shall be blocked; or

3. The correctness of the control rod withdrawal sequence input to the RWM computer shall be verified after loading the sequence.

   Prior to the start of control rod withdrawal towards criticality, the capability of the rod worth minimizer to properly fulfill its function shall be verified by the following checks:

   a. The RWM computer online diagnostic test shall be successfully performed.

   b. Proper annunciation of the selection error of one out-of-sequence control rod shall be verified.

   c. The rod block function of the RWM shall be verified by withdrawing the first rod as an out-of-sequence control rod no more than to the block point.

4. Prior to control rod withdrawal for startup or during refueling, verify that at least two source range channels have an observed count rate of at least three counts per second.

5. When a limiting control rod pattern exists, an instrument functional test of the RBM shall be performed prior to withdrawal of the designated rod(s) and daily thereafter.

c. the operating power level shall be limited so that the MCPR will remain above the MCPR fuel cladding integrity safety limit assuming a single error that results in complete withdrawal of any single operable control rod.

C. Scram Insertion Times

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

% Inserted From Fully Withdrawn	*Average Scram Insertion Times (sec)*
5	0.375
20	0.900
50	2.00
90	3.50

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

% Inserted From Fully Withdrawn	*Average Scram Times (sec)*
5	0.398
20	0.954
50	2.12
90	3.80

2. The maximum scram insertion time for 90% insertion of any operable control rods shall not exceed 7 seconds.

3. If Specification 3.3.C.1 cannot be met, the reactor shall not be made supercritical; if operating, the reactor shall be shut down immediately upon determination that average scram time is deficient.

4. If Specification 3.3.C.2 cannot be met, the deficient control rod shall be con-

C. Scram Insertion Times

1. After refueling outage and prior to operation above 30% power, with reactor pressure above 800 psig, all control rods shall be subject to scram-time measurements from the fully withdrawn position. The scram times shall be measured without reliance on the control rod drive pumps.

2. Following a controlled shutdown of the reactor, but not more frequently than 16 weeks nor less frequently than 32-week intervals, 50% of the control rod drives in each quadrant of the reactor core shall be measured for the scram times specified in Specification 3.3.C. All control rod drives shall have experienced scram test measurements each year. Whenever all of the control rod drive scram times have been measured, an evaluation shall be made to

B. Control Rod Withdrawal

1. Control rod dropout accidents as discussed in Reference 1 can lead to significant core damage. If coupling integrity is maintained, the possibility of a rod dropout accident is eliminated. The overtravel position feature provides a positive check, as only uncoupled drives may reach this position.

Neutron instrumentation response to rod movement provides a verification that the rod is following its drive. Absence of such response to drive movement would indicate an uncoupled condition.

2. 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 Section 6.6.1, and the design evaluation is given in Section 6.6.3 of the SAR. 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. Additionally, the support is not required if all control rods are fully inserted or if an adequate shutdown margin with one control rod withdrawn has been demonstrated, since the reactor would remain subcritical even in the event of complete ejection of the strongest control rod.

3. Control rod withdrawal and insertion sequences are established to assure that the maximum insequence individual control rod or control rod segments which are withdrawn could not be worth enough to cause the rod drop accident design limit of 280 cal/gm to be exceeded if they were to drop out of the core in the manner defined for the rod drop accident. These sequences are developed prior to initial operation of the unit following any refueling outage and the requirement that an operator follow these sequences is supervised by the RWM or a second qualified station employee. These sequences are developed to limit reactivity worths of control rods and

together with the integral rod velocity limiters and the action of the control rod drive system, limits potential reactivity insertion such that the results of a control rod drop accident will not exceed a maximum fuel energy content of 280 cal/gm. The peak fuel enthalpy of 280 cal/gm is below the energy content at which rapid fuel dispersal and primary system damage have been found to occur based on experimental data as is discussed in Reference 2.

The analysis of the control rod drop accident was originally presented in Sections 7.9.3, 14.2.1.2 and 14.2.1.4 of the SAR. Improvements in analytical capability have allowed a more refined analysis of the control rod drop accident.

These techniques are described in a topical report (Reference 2) and two supplements (References 3 and 4). In addition, a banked position withdrawal sequence described in Reference 5 has been developed to further reduce incremental rod worths. Method and basis for the rod drop accident analyses are documented in Reference 1.

By using the analytical models described in those reports coupled with conservative or worst-case input parameters, it has been determined that for power levels less than 20% of rated power, the specified limit on insequence control rod or control rod segment worths will limit the peak fuel enthalpy to less than 280 cal/g. Above 20% power even single operator errors cannot result in out-of-sequence control rod worths which are sufficient to reach a peak fuel enthalpy of 280 cal/g should a postulated control rod drop accident occur.

The following parameters and worst-case assumptions have been utilized in the analysis to determine compliance with the 280 cal/gm peak fuel enthalpy. Each core reload will be analyzed to show conformance to the limiting parameters.

a. an interassembly local peaking factor (Reference 6).

b. the delayed neutron fraction chosen for the bounding reactivity curve

c. a beginning-of-life Doppler reactivity feedback

d. scram times slower than the Technical Specification rod scram insertion rate (Section 3.3.C.1)

e. the maximum possible rod drop velocity of 3.11 fps

f. the design accident and scram reactivity shape function, and

g. the moderator temperature at which criticality occurs

In most cases the worth of insequence rods or rod segments in conjunction with the actual values of the other important accident analysis parameters described above, would most likely result in a peak fuel enthalpy substantially less than 280 cal/g design limit.

Should a control drop accident result in a peak fuel energy content of 280 cal/g, fewer than 660 (7 x 7) fuel rods are conservatively estimated to perforate. This would result in an offsite dose well below the guideline value of 10 CFR 100. For 8 x 8 fuel, fewer than 850 rods are conservatively estimated to perforate, with nearly the same consequences as for the 7 x 7 fuel case because of the rod power differences.

The rod worth minimizer 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 (reference SAR Section 7.9). It serves as a backup to procedural control of control rod worth. In the event that the rod worth minimizer is out of service when required, a licensed operator or other qualified technical employee can manually fulfill the control rod pattern conformance function of the rod worth minimizer. In this case, the normal procedural controls are backed up by independent procedural controls to assure conformance.

4. 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. This is needed for knowledgeable and efficient reactor startup at low neutron levels. 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 from 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 is provided as an added conservatism.

5. The rod block monitor (RBM) is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power operation. Two channels are provided, and one of these may be bypassed from the console for maintenance and/or testing. Tripping of one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage. This system backs up the operator, who withdraws control rods according to a written sequence. The specified restrictions with one channel out of service conservatively assure that fuel damage will not occur due to rod withdrawal errors when this condition exists. During reactor operation with certain limiting control rod patterns, the withdrawal of a designated single control rod could result in one or more fuel rods with MCPR's less than the MCPR fuel cladding integrity safety limit. During use of such patterns it is judged that testing of the RBM system to assure its operability prior to withdrawal of such rods will assure that improper withdrawal does not occur. It is the responsibility of the Nuclear Engineer to identify these limiting patterns and the designated rods either when the patterns are initially established or as they develop due to the occurrence of inoperable control rods in other than limiting patterns.

C. Scram Insertion Times

The control rod system is analyzed 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 fuel cladding integrity safety limit.

Analysis of the limiting power transient shows that the negative reactivity rates resulting from the scram with the average response of all the drives as given in the above specification, provide the required protection, and MCPR remains greater than the fuel cladding integrity safety limit. The mimimum amount of reactivity to be inserted during a scram is controlled by permitting no more than 10% of the operable rods to have long scram times. In the analytical treatment of the transients, 290 milliseconds are allowed between a neutron sensor reaching the scram point and the start of motion of the control rods. This is adequate and conservative when compared to the typically observed time delay of about 210 milliseconds. Approximately 90 milliseconds after neutron flux reaches the trip point, the pilot scram valve solenoid deenergizes and 120 milliseconds later the control rod motion is estimated to actually begin. However, 200 milliseconds rather than 120 milliseconds is conservatively assumed for this time interval in the transient analyses and is also included in the allowable scram insertion times specified in Specification 3.3.C. The scram times for all control rods will be determined at the time of each refueling outage. A representative sample of control rods will be tested following a shutdown. Scram times of new drives are approximately 2.5 to 3 seconds; lower rates of change in scram times following initial plant operation at power are expected. The test schedule provides reasonable assurance of detection of slow drives before system deterioration beyond limits of Specification 3.3.C. The program was developed on the basis of the statistical approach outlined below and judgment.

The history of drive performance accumulated to date indicates that the 90% insertion times of new and overhauled drives approximate a normal distribution about the mean which tends to become skewed toward longer scram times as operating time is accumulated. The probability of a drive not exceeding the mean 90% insertion time by 0.75 seconds is greater than 0.999 for a normal distribution. The measurement of the scram performance of the drives surrounding a drive exceeding the expected range of scram performance will detect local variations and also provide assurance that local scram time limits are not exceeded. Continued monitoring of other drives exceeding the expected range of scram times provides surveillance of possible anomalous performance.

The numerical values assigned to the predicted scram performance are based on the analysis of the Dresden 2 startup data and of data from other BWR's such as Nine Mile Point and Oyster Creek.

The occurrence of scram times within the limits, but significantly longer than average, should be viewed as an indication of a systematic problem with control rod drives, especially if the number of drives exhibiting such scram times exceeds eight, the allowable number of inoperable rods.

The occurrence of scram times within the limits, but significantly longer than average, should be viewed as an indication of a systematic problem with control rod drives, especially if the number of drives exhibiting such scram times exceeds eight, the allowable number of inoperable rods.

D. Control Rod Accumlators

The basis for this specification was not described in the SAR and is therefore presented in its entirety. Requiring no more than one inoperable accumulator in any nine-rod square array is based on a series of XY PDQ-4 quarter core calculations of a cold clean core. The worst case in a nine-rod withdrawal sequence resulted in a $k_{eff} < 1.0$. Other repeating rod sequences with more rods withdrawn resulted in $k_{eff} > 1.0$. At reactor pressures in excess of 800 psig, even those control rods with inoperable accumulators will be able to meet required scram insertion times due to the action of reactor pressure. In addition, they may be normally inserted using the control rod drive hydraulic system. Procedural control will assure that control rods with inoperable accumulators will be spaced in a one-in-nine array rather than grouped together.

E. Reactivity Anomalies

During each fuel cycle, excess operating 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 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% Δk. Deviations in core reactivity greater than 1% Δk are not expected and require thorough evaluation. A 1% 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. Economic Generation Control System

Operation of the facility with the economic generation control system (EGC) (automatic flow control) is limited to the range of 65% to 100% of rated core flow. In this flow range and with reactor power above 20%, the reactor could safely tolerate a rate of change of load of 8 MWe/sec (reference SAR Section 7.3.5).

Limits within the EGC and the flow control system prevent rates of change greater than approximately 4 MWe/sec. When EGC is in operation, this fact will be indicated on the main control room console. The results of initial testing will be provided to the NRC before the onset of routine operation with EGC.

References

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

2. C. J. Paone, R. C. Stirn, and J. A. Wooley, 'Rod Drop Accident Analysis for Large BWR's', GE Topical Report NEDO-10527, March 1972.

3. C. J. Paone, R. C. Stirn, and R. M. Young, 'Rod Drop Accident Analysis for Large BWR's, Supplement 1, GE Topical Report NEDO-10527, July 1972.

4. J. M. Haun, C. J. Paone, and R. C. Stirn, 'Rod Drop Accident Analysis for Large BWR's, Addendum 2, Exposed Cores,' Supplement 2, GE Topical Report NEDO-10527, January 1973.

5. C. J. Paone, "Banked position withdrawal sequence," Licensing topical Report NEDO-21231, January, 1977.

6. To include the power spike effect caused by gaps between fuel pellets.

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

3.4 LIMITING CONDITIONS FOR OPERATION BASES

A. The design objective of the standby liquid control system is to provide the capability of bringing the reactor from full power to a cold, xenon-free shutdown 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 which produces a concentration of no less than 600 ppm of boron in the reactor core in approximately 90 to 120 minutes with imperfect mixing. A boron concentration of 600 ppm in the reactor core is required to bring the reactor from full power to 3% Δk or more subcritical condition considering the hot to cold reactivity swing, xenon poisoning and an additional margin in the reactor core for imperfect mixing of the chemical solution in the reactor water. A normal quantity of 3470 gallons of solution having a 13.4% sodium pentaborate concentration is required to meet this shutdown requirement.

The time requirement (90 to 120 minutes) for insertion of the boron solution was selected to override the rate of reactivity insertion due to cooldown of the reactor following the xenon poison peak. For a required pumping rate of 39 gpm, the maximum storage volume of the boron solution is established as 4875 gallons (195 gallons are contained below the pump suction and, therefore, cannot be inserted).

Boron concentration, solution temperature, and volume are checked on a frequency to assure a high reliability of operation of the system should it ever be required. Experience with pump operability indicates that monthly testing is adequate to detect if failures have occurred.

The only practical time to test the standby liquid control system is during a refueling outage and by initiation from local stations. Components of the system are checked periodically as described above and make a functional test of the entire system on a frequency of less than once each refueling outage unnecessary. A test of explosive charges from one manufacturing batch is made to assure that the charges are satisfactory. A continual check of the firing circuit continuity is provided by pilot lights in the control room.

B. Only one of the two standby liquid control pumping circuits is needed for proper operation of the system. If one pumping circuit is found to be inoperable, there is no immediate threat to shutdown capability, and reactor operation may continue while repairs are being made. Assurance that the remaining system will perform its intended function and that the reliability of the system is good is obtained by demonstrating operation of the pump in the operable circuit at least once daily. A reliability analysis indicates that the plant can be operated safely in this manner for 7 days.

C. The solution saturation temperature of 13% sodium pentaborate, by weight, is 59° F. The solution shall be kept at least 10° F above the saturation temperature to guard against boron precipitation. The 10° F margin is included in Figure 3.3-1. Temperature and liquid level alarms for the system are annunciated in the control room.

Pump operability is checked on a frequency to assure a high reliability of operation of the system should it ever be required.

Once the solution has been made up, boron concentration will not vary unless more boron or more water is added. Level indication and alarm indicate whether the solution volume has changed, which might indicate a possible solution concentration change. Considering these factors, the test interval has been established.

is being done which has the potential for draining the reactor vessel.

3. When irradiated fuel is in the reactor and the vessel head is removed, the suppression chamber may be drained completely and no more than one control rod drive housing opened at any one time provided that the spent fuel pool gate is open and the fuel pool water level is maintained at a level of greater than 33 feet above the bottom of the pool. Additionally, a minimum condensate storage reserve of 230,000 gallons shall be maintained, no work shall be performed in the reactor vessel while a control rod drive housing is blanked following removal of the control rod drive, and a special flange shall be available which can be used to blank an open housing in the event of a leak.

4. When irradiated fuel is in the reactor and the vessel head is removed, work that has the potential for draining the vessel may be carried on with less than 112,200 ft^3 of water in the suppression pool, provided that: (1) the total volume of water in the suppression pool, refueling cavity, and the fuel storage pool above the bottom of the fuel pool gate is greater than 112,200 ft^3; (2) the fuel storage pool gate is removed; (3) the low-pressure core and containment cooling systems are operable; and (4) the automatic mode of the drywell sump pumps is disabled.

G. **Maintenance of Filled Discharge Pipe**

1. Whenever core spray, LPCI mode of the RHR, HPCI, or RCIC are required to be operable, the discharge piping from the pump discharge of these systems to the last check valves shall be filled.

G. **Maintenance of Filled Discharge Pipe**

The following surveillance requirements shall be adhered to to assure that the discharge piping of the core spray, LPCI mode of the RHR, HPCI, and RCIC are filled:

1. Every month prior to the testing of the LPCI mode of the RHR and core spray ECCS, the discharge piping of these systems shall be vented from the high point and water flow observed.

cycle by assuring that water can be run through the drain lines and actuating the air-operated valves by operation of the following sensors:

1) loss of air

2) equipment drain sump high level

3) vault high level

d. The condenser pit 5-foot trip circuits for each channel shall be checked once a month. A logic system functional test shall be performed during each refueling outage.

I. **Average Planar LHGR**

During steady-state power operation, the average linear heat generation rate (APLHGR) of all the rods in any fuel assembly, as a function of average planar exposure, at any axial location, shall not exceed the maximum average planar LHGR shown in Figure 3.5-1. If at any time during operation it is determined by normal surveillance that the limiting value for APLHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the APLHGR is not returned in within the prescribed limits within 2 hours, the reactor shall be brought to the cold shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

J. **Local LHGR**

During steady-state power operation, the linear heat generation rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed the maximum allowable LHGR. If at any time during operation it is determined by normal surveillance that the limiting value for LHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the LHGR is not returned to

I. **Average Planar LHGR**

Daily during steady state operation above 25% rated thermal power, the average planar LHGR shall be determined.

J. **Local LHGR**

Daily during steady-state power operation above 25% of rated thermal power, the local LHGR shall be determined.

within the prescribed limits within 2 hours, the reactor shall be brought to the cold shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

Maximum allowable LHGR for all 8X8 fuel types is 13.4 KW/ft. For 7X7 and mixed oxide fuel, the maximum allowable LHGR is as follows:

$$LHGR_{max} < LHGR_d \left[1 - (\Delta P/P)_{max} (L/L_t) \right]$$

where:

$LHGR_d$ = design LHGR

= 17.5 kW/ft.

$(\Delta P/P)_{max}$ = maximum power spiking penalty

= .035 initial core fuel

= .029 reload 1, 7 x 7 fuel

= .028 reload 1, 7 x 7 mixed oxide fuel

L_t = total core length

= 12 feet

L = Axial distance from bottom of core

K. Minimum Critical Power Ratio (MCPR)

During steady-state operation MCPR shall be greater than or equal to

1.35 (7 x 7 fuel)
1.35 (8 x 8 fuel)

at rated power and flow. If at any time during operation it is determined by normal surveillance that the limiting value for MCPR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the steady-state MCPR is not returned to within the prescribed limits within 2 hours, the reactor shall be brought to the cold shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits. For core flows other than rated, these nominal values of MCPR shall be increased by a factor of k_f where k_f is as shown in Figure 3.5.2.

K. Minimum Critical Power Ratio (MCPR)

The MCPR shall be determined daily during steady-state power operation above 25% of rated thermal power.

3.5 LIMITING CONDITION FOR OPERATION BASES

A. Core Spray and LPCI Mode of the RHR System

This specification assures that adequate emergency cooling capability is available whenever irradiated fuel is in the reactor vessel.

Based on the loss-of-coolant analytical methods described in General Electric Topical Report NEDO-20566 and the specific analysis in Reference 1, core cooling systems provide sufficient cooling to the core to dissipate the energy associated with the loss-of-coolant accident, to limit calculated fuel cladding temperature to less than 2200°F, to assure that core geometry remains intact, to limit cladding metal-water reaction to less than 1%, and to limit the calculated local metal-water reaction to less than 17%.

The limiting conditions of operation in Specifications 3.5.A.1 through 3.5.A.6 specify the combinations of operable subsystems to assure the availability of the minimum cooling systems noted above. No single failure of ECCS equipment occurring during a loss-of-coolant accident under these limiting conditions of operation will result in inadequate cooling of the reactor core.

Core spray distribution has been shown, in full-scale tests of systems similar in design to that of Quad-Cities 1 and 2, to exceed the minimum requirements by at least 25%. In addition, cooling effectiveness has been demonstrated at less than half the rated flow in simulated fuel assemblies with heater rods to duplicate the decay heat characteristics of irradiated fuel. The accident analysis is additional conservative in that no credit is taken for spray cooling of the reactor core before the internal pressure has fallen to 90 psig.

The LPCI mode of the RHR system is designed to provide emergency cooling to the core by flooding in the event of a loss-of-coolant accident. This system functions in combination with the core spray system to prevent excessive fuel cladding temperature. The LPCI mode of the RHR system in combination with the core spray subsystem provides adequate cooling for break areas of approximately 0.2 ft^2 up to and including 4.18 ft^2, the latter being the double-ended recirculation line break with the equalizer line between the recirculation loops closed without assistance from the high-pressure emergency core cooling subsystems.

The allowable repair times are established so that the average risk rate for repair would be no greater than the basic risk rate. The method and concept are described in Reference 3 Using the results developed in this reference, the repair period is found to be less than half the test interval. This assumes that the core spray subsystems and LPCI constitute a one-out-of-two system; however, the combined effect of the two systems to limit excessive cladding temperature must also be considered. The test interval specified in Specification 4.5 was 3 months. Therefore, an allowable repair period which maintains the basic risk considering single failures should be less than 30 days, and this specification is within this period. For multiple failures, a shorter interval is specified; to improve the assurance that the remaining systems will function, a daily test is called for. Although it is recognized that the information given in Reference 1 provides a quantitative method to estimate allowable repair times, the lack of operating data to support the analytical approach prevents complete acceptance of this method at this time. Therefore, the times stated in the specific items were established with due regard to judgment.

Should one core spray subsystem become inoperable, the remaining core spray subsystem and the entire LPCI mode of the RHR system are available should the need for core cooling arise. To assure that the remaining core spray, the LPCI mode of the RHR system, and the diesel generators are available, they are demonstrated to be operable immediately. This demonstration includes a manual initiation of the pumps and associated valves and diesel generators. Based on judgments of the reliability of the remaining systems, i.e., the core spray and LPCI, a 7-day repair period was obtained.

H. Condensate Pump Room Flood Protection

See Specification 3.5.H.

I. Average Planar LHGR

This specification assures that the peak cladding temperature following the postulated design-basis loss-of-coolant accident will not exceed the 2200°F limit specified in the 10 CFR 50 Appendix K considering the postulated effects of fuel pellet densification.

The peak cladding temperature following a postulated loss-of-coolant accident is primarily a function of the average heat-generation rate of all the rods of a fuel assembly at any axial location and is only secondarily dependent 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 cladding temperature by less than ±20°F relative to the peak temperature for a typical fuel design, the limit on the average planar LHGR is sufficient to assure that calculated temperatures are below the limit. The maximum average planar LHGR's shown in Figure 3.5-1 are based on calculations employing the models described in Reference 2.

J. Local LHGR

This specification assures that the maximum linear heat-generation rate in any rod is less than the design linear heat-generation rate even if fuel pellet densification is postulated. The power spike penalty is discussed in Reference 2 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.

K. Minimum Critical Power Ratio (MCPR)

The steady state values for MCPR specified in this specification were selected to provide margin to accommodate transients and uncertainties in monitoring the core operating state as well as uncertainties in the critical power correlation itself. These values also assure that operation will be such that the initial condition assumed for the LOCA analysis plus two percent for uncertainty is satisfied. For any of the special set of transients or disturbances caused by single operator error or single equipment malfunction, it is required that design analyses initialized at this steady-state operating limit yield a MCPR of not less than that specified in Specification 1.1.A at any time during the transient, assuming instrument trip settings given in Specification 2.1. For analysis of the thermal consequences of these transients, the value of MCPR stated in this specification for the limiting condition of operation bounds the initial value of MCPR assumed to exist prior to the initiation of the transients. This initial condition, which is used in the transient analyses, will preclude violation of the fuel cladding integrity safety limit. Assumptions and methods used in calculating the required steady state MCPR limit for each reload cycle are documented in Reference 2. The results apply with increased conservatism while operating with MCPR's greater than specified.

The most limiting transients with respect to MCPR are generally:

a) Rod withdrawal error

b) Load rejection or Turbine Trip without bypass

c) Loss of feedwater heater

Several factors influence which of these transients results in the largest reduction in critical power ratio such as the specific fuel loading, exposure, and fuel type. The current cycles reload licensing analyses specifies the limiting transients for a given exposure increment for each fuel type. The values specified as the Limiting Condition of Operation are conservatively chosen to bound the most restrictive over the entire cycle for each fuel type.

For core flow rates less than rated, the steady state MCPR is increased by the formula given in the specification. This assures that the MCPR will be maintained greater than that specified in Specification 1.1.A even in the event that the motor-generator set speed controller causes the scoop tube positioner for the fluid coupler to move to the maximum speed position.

References

1. "Loss-of-Coolant Analysis Report for Dresden Units 2, 3 and Quad Cities Units 1, 2 Nuclear Power Stations," NEDO-24146A*, April, 1979

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

3. I. M. Jacobs and P. W. Marriott, GE Topical Report APED 5736, "Guidelines for Determining Safe Test Intervals and Repair Times for Engineered Safeguards," April, 1969.

* Approved revision at time of plant operation.

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

Should the switches at levels (a) and (b) fail or the operator fail to trip the circulating water pumps on alarm at level (b), the actuation of either level switch pair at level (c) shall trip the circulating water pumps automatically and alarm in the control room. These redundant level switch pairs at level (c) are designed and installed to IEEE 279, 'Criteria for Nuclear Power Plant Protection Systems.' As the circulating water pumps are tripped, either manually or automatically, at level (c) of 5 feet, the maximum water level reached in the condenser pit due to pumping will be at elevation 568 feet 6 inches elevation (10 feet above condenser pit floor elevation 558 feet 6 inches; 5 feet plus an additional 5 feet attributed to pump coastdown).

In order to prevent the RHR service water pump motors and diesel-generator cooling water pump motors from overheating, a vault cooler is supplied for each pump. Each vault cooler is designed to maintain the vault at a maximum 105° F temperature during operation of its respective pump. For example, if diesel generator cooling water pump 1/2-3903 starts, its cooler also starts and maintains the vault at 105° F by removing heat supplied to the vault by the motor of pump 1/2-3903. If, at the same time that pump 1/2-3903 is in operation, RHR service water pump 1C starts, its cooler will also start and compensate for the added heat supplied to the vault by the 1C pump motor keeping the vault at 105° F.

Each of the coolers is supplied with cooling water from its respective pump's discharge line. After the water has been passed through the cooler it returns to its respective pump's suction line. In this way the vault coolers are supplied with cooling water totally inside the vault. The cooling water quantity needed for each cooler is approximately 1% to 5% of the design flow of the pumps so that the recirculation of this small amount of heated water will not affect pump or cooler operation.

Operation of the fans and coolers is required during shutdown and thus additional surveillance is not required.

Watertight vaults for the ECCS pumps in the reactor building are tested in essentially the same manner and frequency as described for the condenser pump room vaults.

Verification that access doors to each vault are closed following entrance by personnel is covered by station operating procedures.

The LHGR shall be checked daily to determine if fuel burnup or control rod movement has caused changes in power distribution. Since changes due to burnup are slow and only a few control rods are moved daily, a daily check of power distribution is adequate.

Average Planar LHGR

At core thermal power levels less than or equal to 25%, operating plant experience and thermal hydraulic analyses indicate that the resulting average planar LHGR is below the maximum average planar LHGR by a considerable margin; therefore, evaluation of the average planar LHGR below this power level is not necessary. The daily requirement for calculating average planar LHGR above 25% rated thermal power is sufficient, since power distribution shifts are slow when there have not been significant power or control rod changes.

Local LHGR

The LHGR as a function of core height shall be checked daily during reactor operation at greater than or equal to 25% power to determine if fuel burnup or control rod movement has caused changes in power distribution. A limiting LHGR value is precluded by a considerable margin when employing any permissible control rod pattern below 25% rated thermal power.

Minimum Critical Power Ratio (MCPR)

At core thermal power levels less than or equal to 25%, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience and thermal hydraulic analysis indicate that the resulting MCPR value is in excess of requirements by a considerable margin. With this low void content, any inadvertent core flow increase would only place operation in a more conservative mode relative to MCPR.

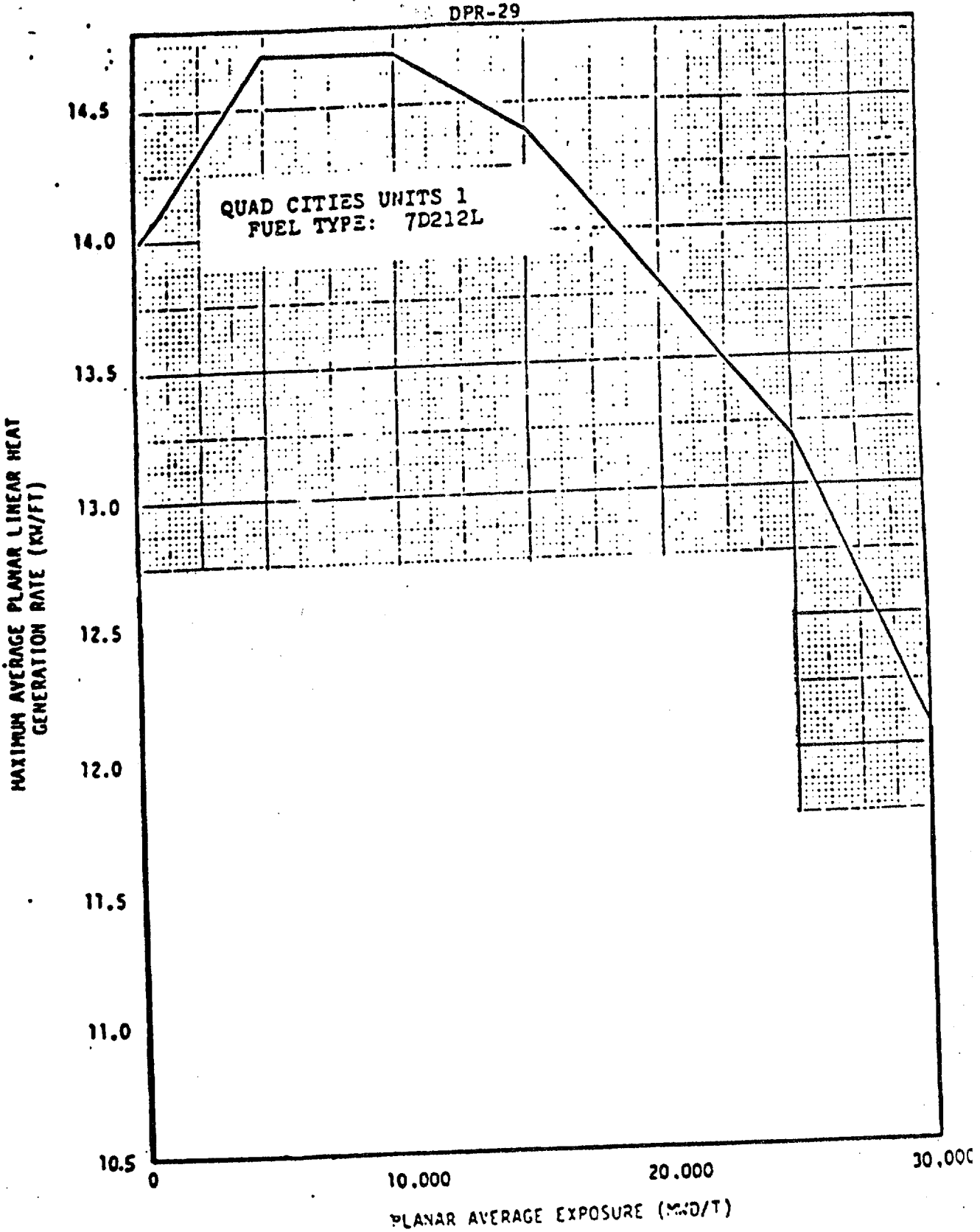
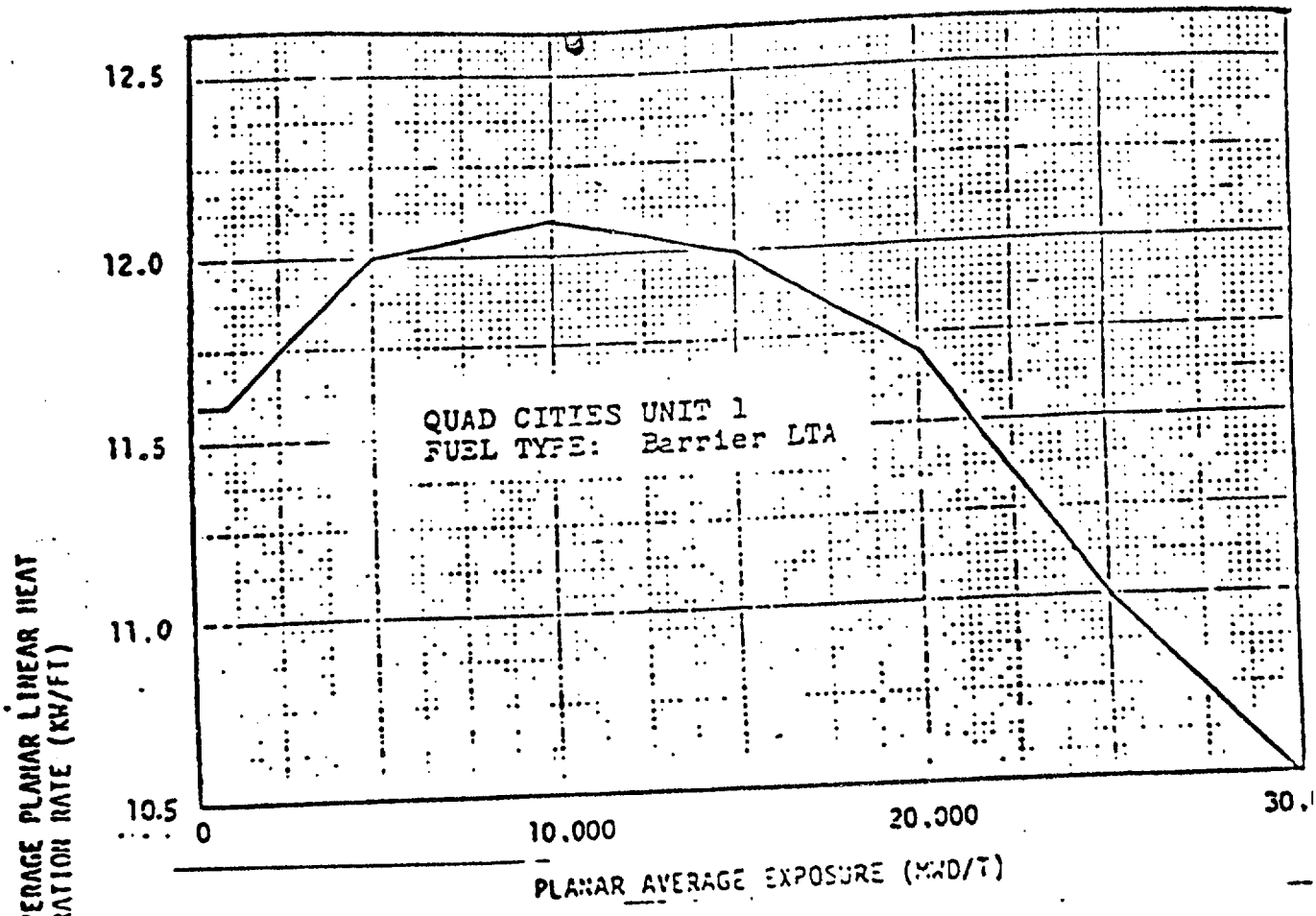


FIGURE 3.5-1
(Sheet 1 of 6)

MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VS. PLANAR AVERAGE EXPOSURE



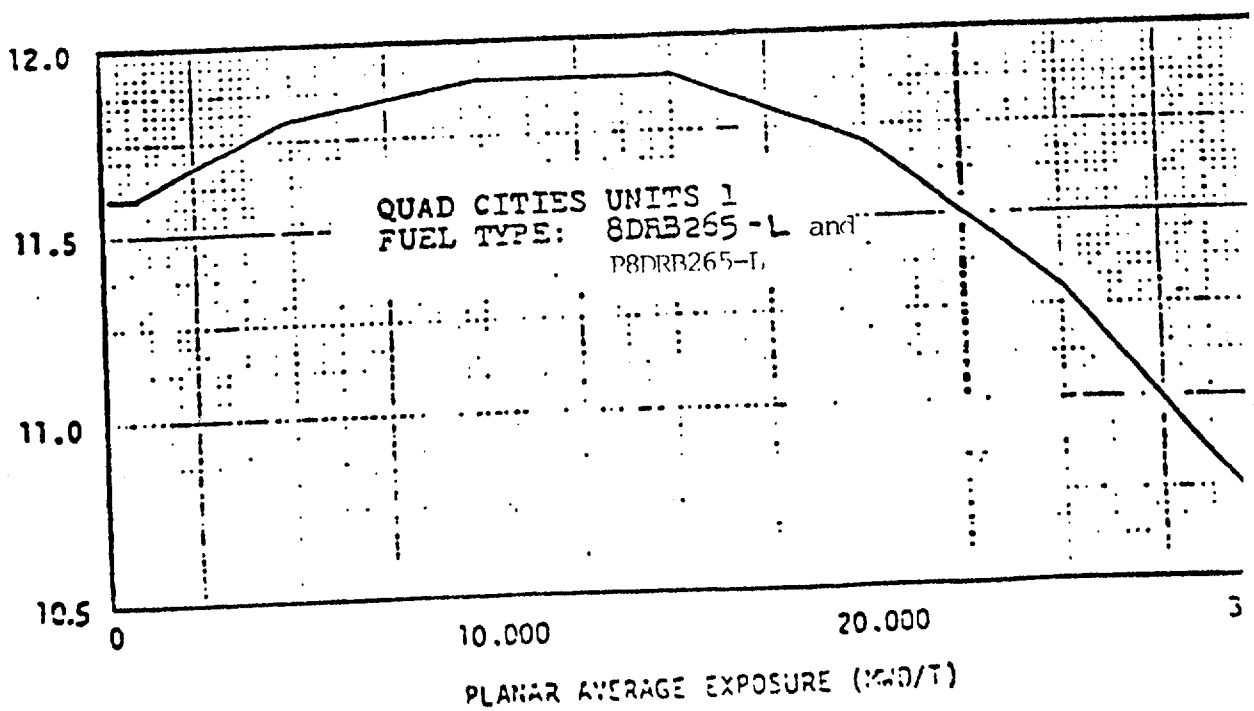
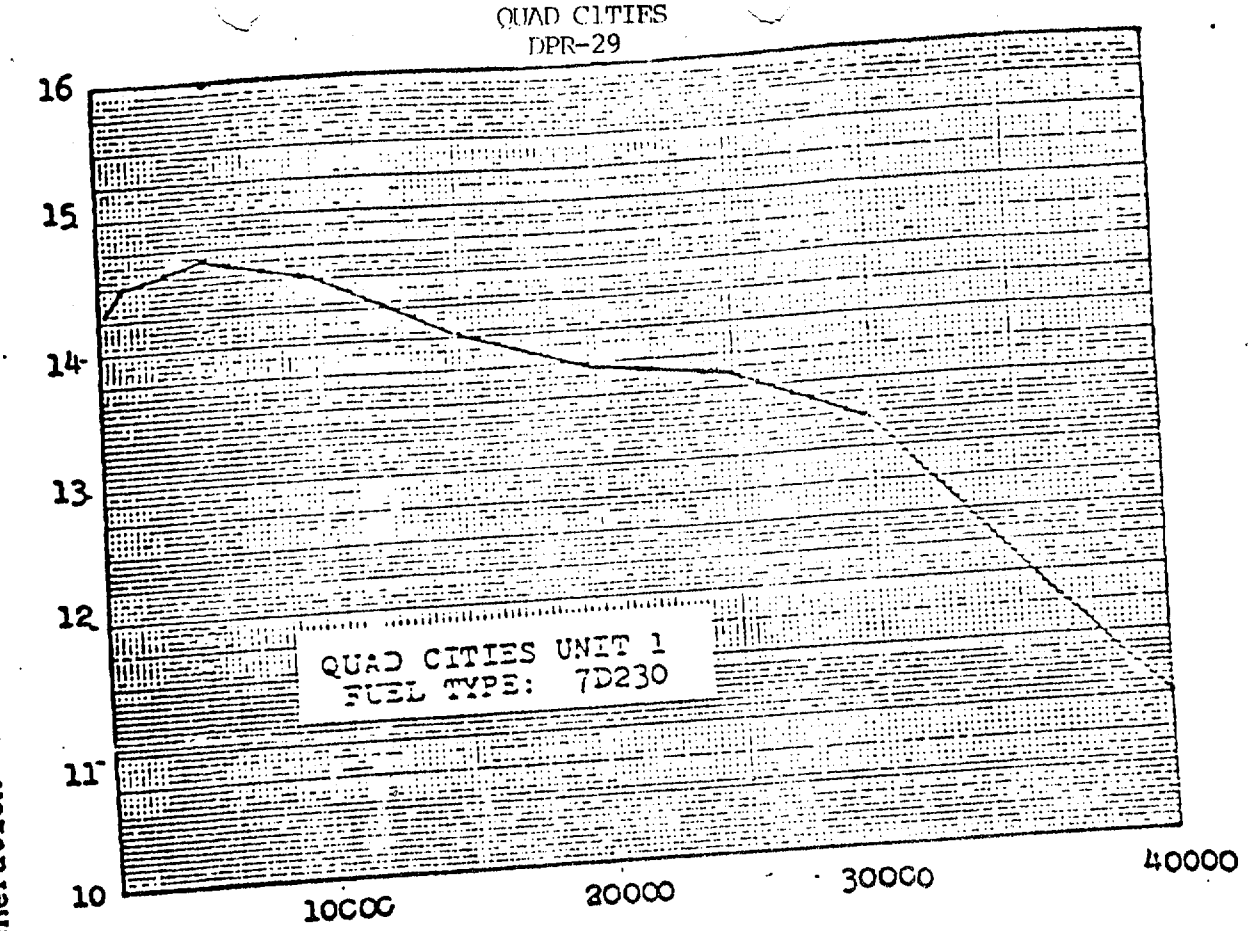


FIGURE 3.5-1
(Sheet 2 of 6)

MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VS. PLANAR AVERAGE EXPOSURE



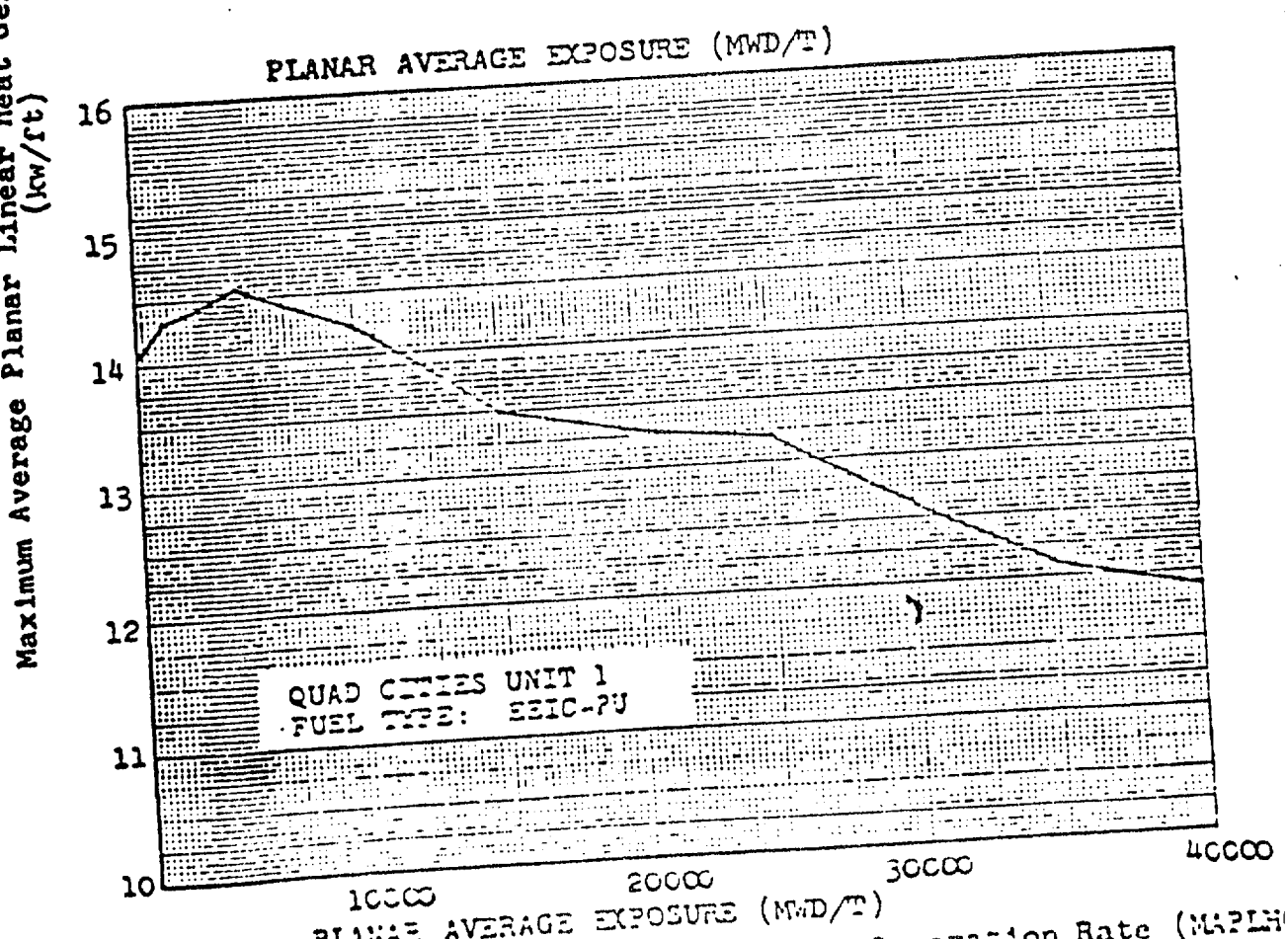
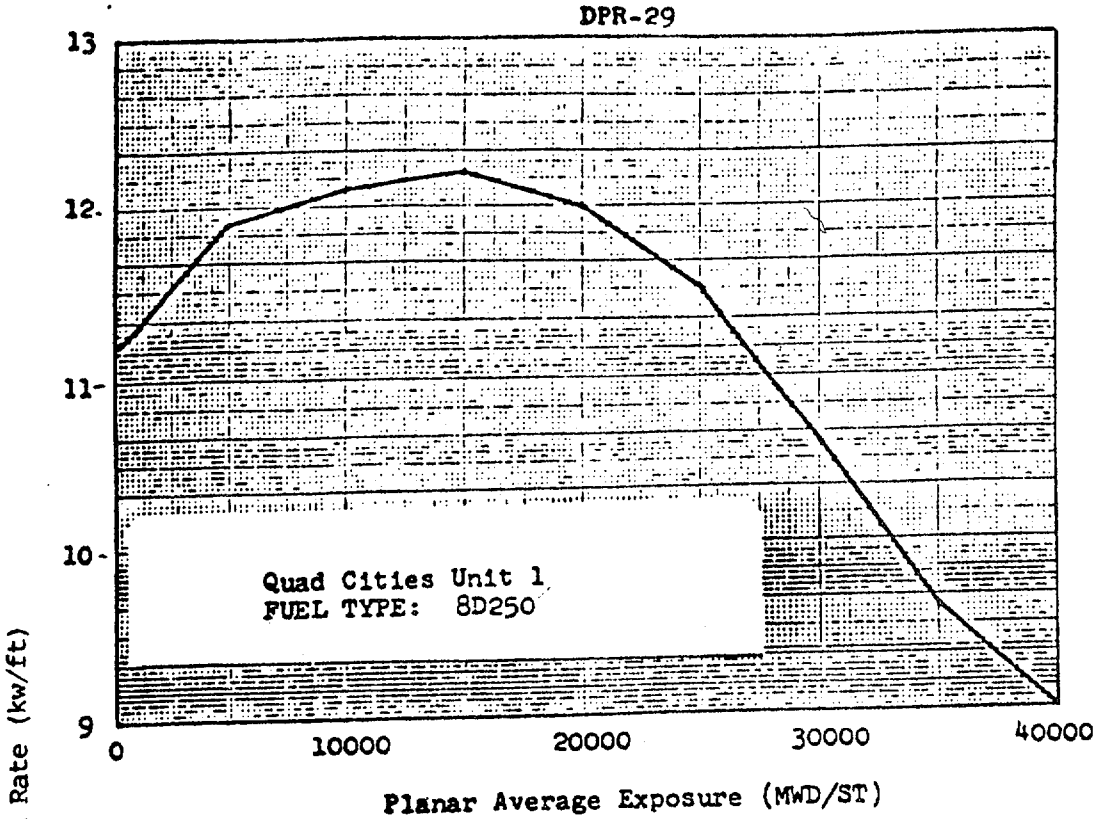


Figure 3.5-1 Maximum Average Planar Linear Heat Generation Rate (MAPLH
(Sheet 3 of 6) vs. Planar Average Exposure



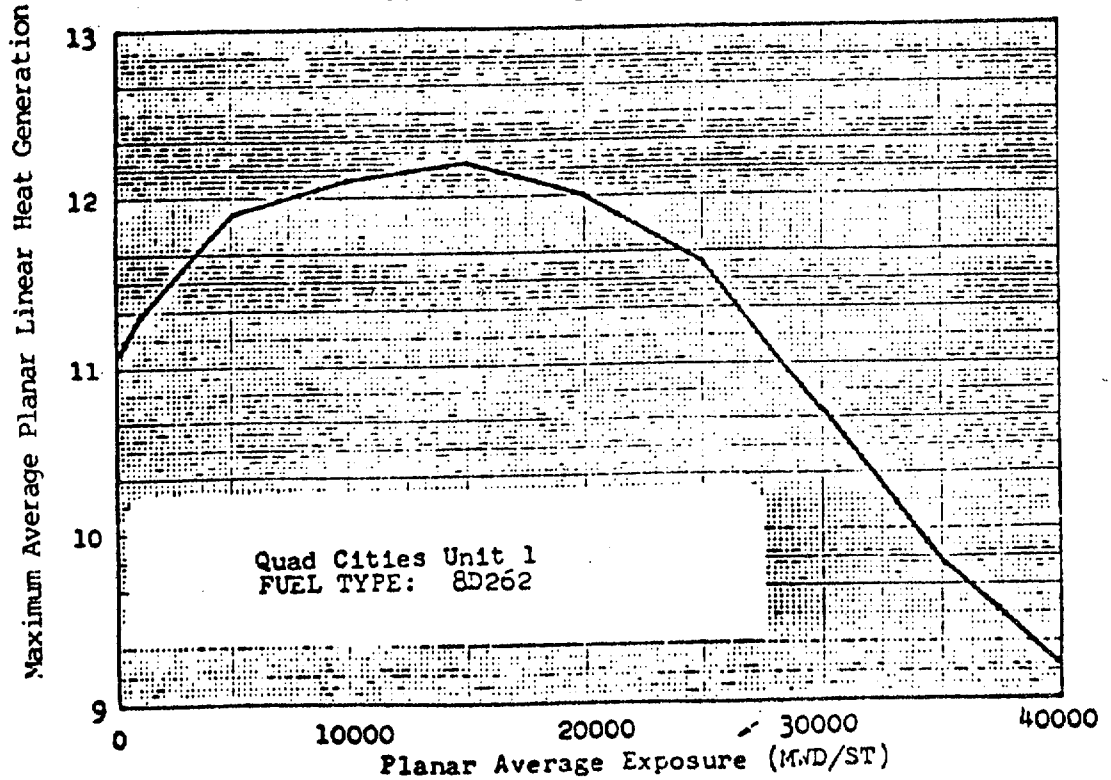
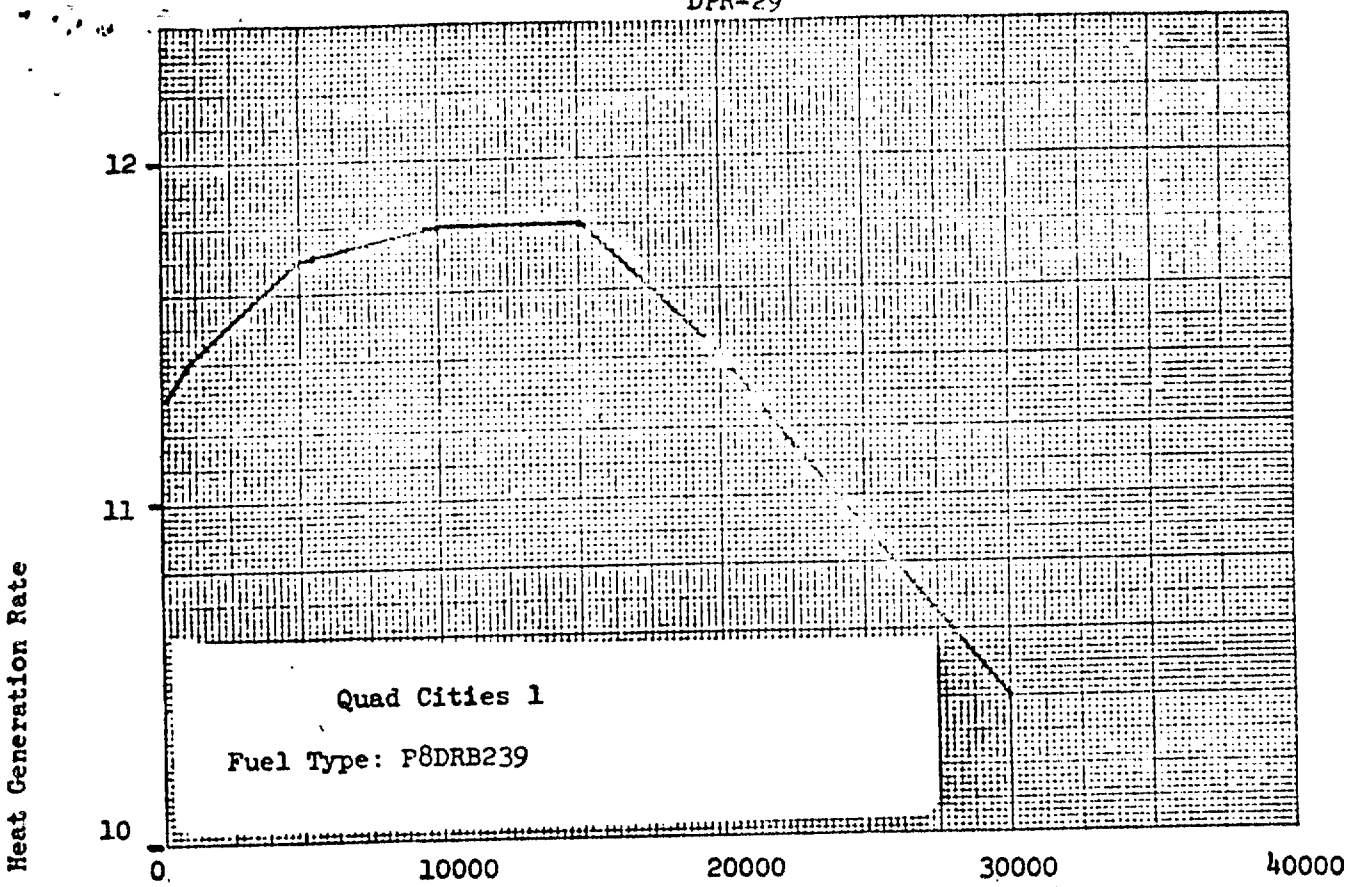


Figure 3.5-1 (Sheet 4 of 6) Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) vs. Planar Average Exposure

Quad Cities
DPR-29



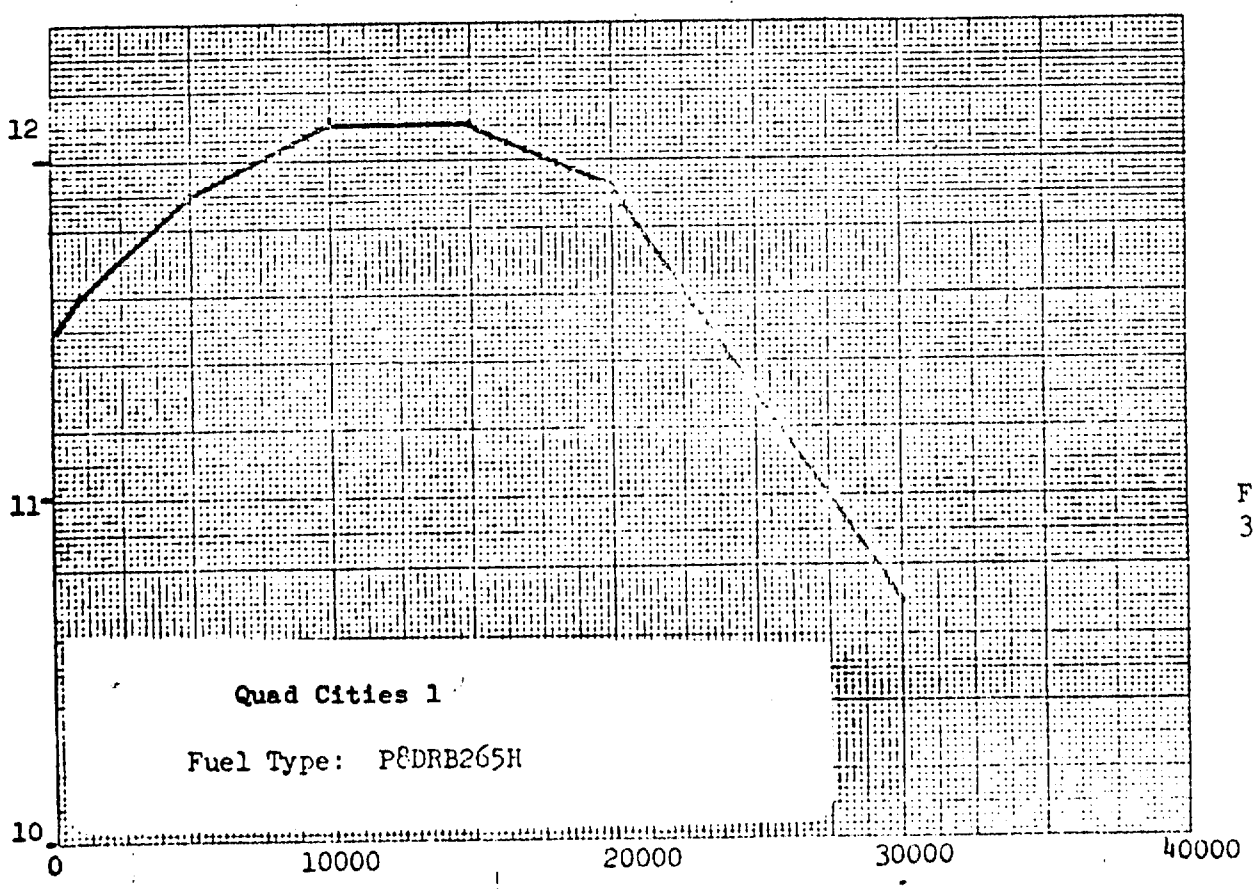
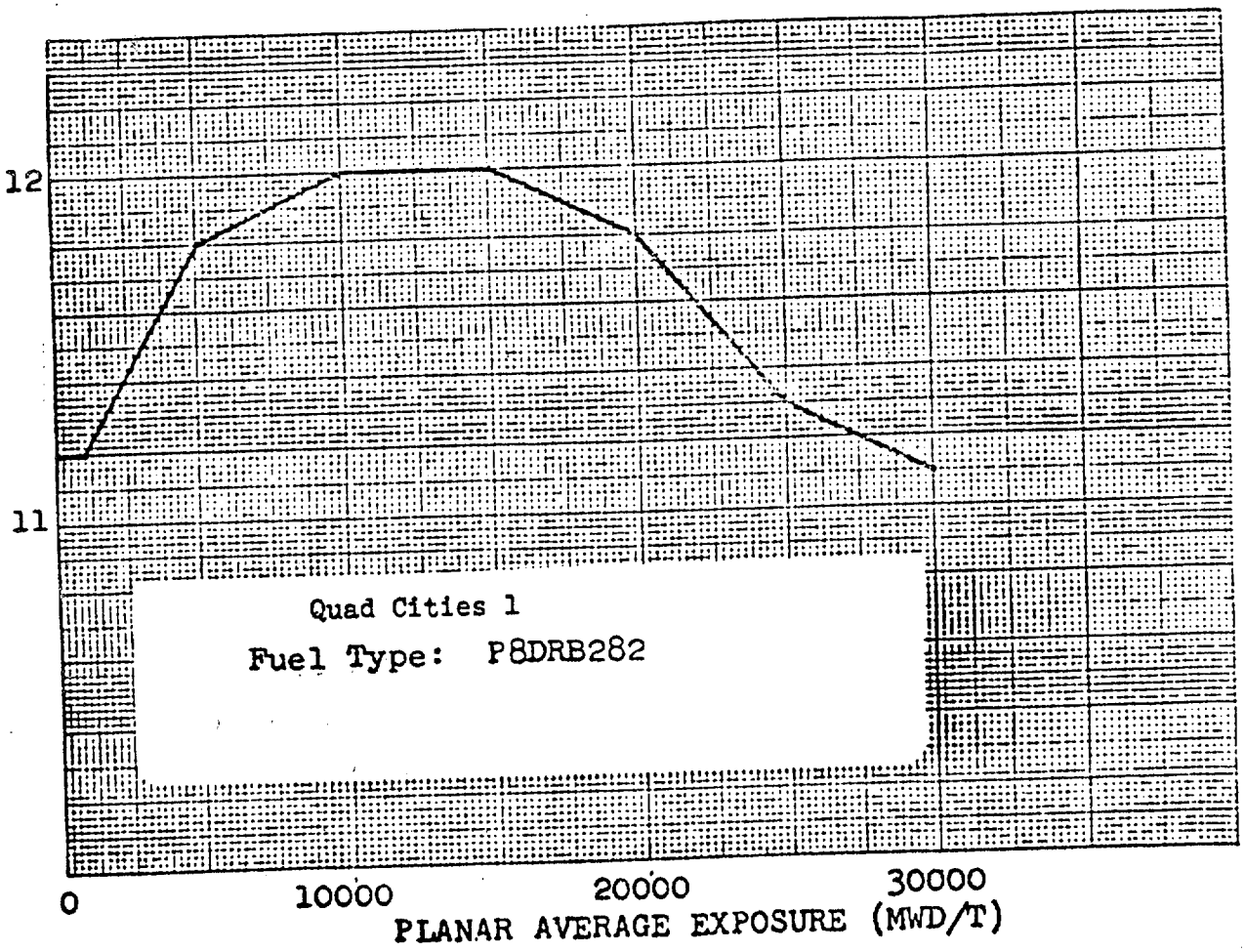


FIGURE 3.5-1

Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) vs. Planar Average Exposure (Sheet 5 of 6)



Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) vs. Planar Average Exposure

(Sheet 6 of 6)

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 61 TO FACILITY OPERATING LICENSE NO. DPR-29

COMMONWEALTH EDISON COMPANY

AND

IOWA-ILLINOIS GAS AND ELECTRIC COMPANY

QUAD CITIES NUCLEAR POWER STATION, UNIT NO. 1
DOCKET NO. 50-254

Introduction

By letter dated September 2, 1980 (Ref. 1), and supplemented by letter dated October 3, 1980 (Ref. 2), Commonwealth Edison Company (CECo or the licensee), proposed an amendment to Quad Cities Unit 1 Appendix A, Technical Specifications. CECo has proposed the amendment to support its review of future reloads for Quad Cities Unit 1 under the provisions of 10 CFR 50.59.

Our approval is only for the proposed amendment and does not constitute approval of future reloads under the provisions of 10 CFR 50.59.

Evaluation

Safety Limit Minimum Critical Power Ratio (SLMCPR)

This change provides SLMCPR values in the Technical Specifications for all currently approved core loadings. With retrofit 8x8 fuel in the core the SLMCPR limit is specified as 1.07. Without retrofit 8x8 fuel, the SLMCPR is 1.06. These limits have previously been found to be acceptable for this use in Reference 3 and on this basis the proposed change is acceptable.

Rod Drop Accident (RDA) Design Limit

The RDA design limit has been modified from 1.3%Δ maximum rod worth to 280 cal/gm peak fuel enthalpy rise. The 280 cal/gm design limit is acceptable per Standard Review Plan NUREG 75-087. Also, the power level below which the rod worth minimizer is required was increased from 10% to 20% of rated power. This is conservative by comparison to the previous specification, is consistent with reactor safety analyses, and is acceptable.

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Maximum Average Planar Linear Heat Generation Rate (MAPLHGR)

New MAPLHGR curves reflecting the improved flooding characteristics of retrofit 8x8 fuel have been proposed by the licensee. Curves for 8x8, 8x8 retrofit, and 7x7 fuel of the various enrichments anticipated for future Quad Cities 1 reloads and extending to burnups of 40,000 MWd/t have been proposed (References 1 and 4).

The new curves are based on an assumed fuel loading with 156 retrofit assemblies. Any reload with fewer such assemblies will be nonconservative with respect to the analyzed case and therefore outside the scope of this approval.

Based on our previous approval of MAPLHGR curves reflecting 8x8 retrofit fuel reflood characteristics (Reference 5) and extension of burnup to 40,000 MWd/t (Reference 6), the licensee's proposed changes are acceptable.

Power Peaking

The licensee has proposed to adjust the Average Power Range Monitor (APRM) amplifier gain based on the Maximum Fraction of Limiting Power Density (MFLPD). Such an adjustment would be made in the event of operation with a MFLPD greater than the Fraction of Rated Power (FRP), with the objective of preventing the fuel cladding integrity safety limits from being exceeded during anticipated operational transients. This adjustment will be applied above 25% rated thermal power which is consistent with the LHGR surveillance requirements and the Standard Technical Specifications.

Previously this objective has been met by reducing the APRM trip settings through multiplication by the ratio of the Limiting Total Peaking Factor (LTPF) to the Total Peaking Factor (TPF). Such a reduction in set points is required in the event of operation with TPF>LTPF.

We have concluded that the maximum reactor power which could be attained during anticipated operational transients with the proposed APRM gain adjustment would be no greater than would be attained with the current procedure for adjusting APRM setpoints. This conclusion is based on the equivalence of the ratio FRP/MFLPD to the ratio LTPF/TPF, and can be explained as follows.

The LTPF can be expressed as the design linear heat generation rate divided by the plant rated thermal power per unit length of fuel rod. In a similar manner the TPF can be expressed as the maximum linear heat generation rate divided by the plant operating power per unit length of fuel rod. From these definitions it is easily determined that the ratio LTPF/TPF is the ratio of the design linear heat generation rate to the maximum linear heat generation rate times the fraction of rated thermal power, or 1/MFLPD*FRP. Thus FRP/MFLPD and LTPF/TPF are equivalent.

However, instead of multiplying the APRM set points by FRP/MFLPD the same result can be achieved by multiplying the APRM reading by MFLPD/FRP to get a gain-adjusted APRM reading. If the reactor is operating in a steady state mode the APRM reading (before gain adjustment) is equal to FRP. Therefore by adjusting the gain until the APRM reading is equal to MFLPD, the APRM reading has effectively been multiplied by MFLPD/FRP as required.

To summarize, the proposed formulation does not involve a reduction in margin to the trip point, and eliminates the need for different limits for different fuel types. In addition adjusting the APRM gain is much easier than changing the APRM trip setting, so that there is less chance for human error.

Reactor Protection System (RPS) Delay Time

The licensee has proposed to change the RPS delay time from 100 to 50 msec (time from opening of the sensor contact up to and including the opening of the trip actuator contacts). This change stems from an inconsistency which has existed between the Technical Specification value of 100 msec and the 50 msec value assumed by General Electric in the licensing analysis. The licensee has confirmed that the procedures used for determining RPS delay time are consistent with the General Electric use and definition of a 50 msec delay time in the licensing analysis. The staff has confirmed that the licensee has in place the capability for demonstrating compliance with the more restrictive specification. The proposed change is acceptable.

Typographical Corrections and Clarification of Bases

The remaining changes fall into the category of typographical corrections and clarification of bases and do not, as such, represent a significant safety concern.

Environmental Consideration

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and pursuant to 10 CFR Section 51.5(d)(4) that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment

does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: December 5, 1980

References

1. Letter from R. F. Janecek (CECo) to Director of Nuclear Reactor Regulation (USNRC), dated September 2, 1980.

2. Letter from R. F. Janecek (CECo) to Director of Division of Licensing (USNRC), dated October 3, 1980.

3. Letter from D. G. Eisenhut (USNRC) to R. Gridley (GE) dated May 12, 1978.

4. "Loss-of-Coolant Accident Analysis Report for Dresden Units 2, 3, and Quad Cities Units 1, 2 Nuclear Power Station," NEDO-24146A, dated April 1979.

5. Letter from D. L. Ziemann (NRC) to Cordell Reed (CECo), dated April 24, 1979.

6. Letter from T. A. Ippolito (NRC) to D. L. Peoples (CECo), dated December 28, 1979.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-254

COMMONWEALTH EDISON COMPANY

AND

IOWA-ILLINOIS GAS AND ELECTRIC COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 61 to Facility Operating License DPR-29 issued to Commonwealth Edison Company and Iowa-Illinois Gas and Electric Company, which revised the Technical Specifications for operation of the Quad Cities Nuclear Power Station, Unit No. 1, located in Rock Island County, Illinois. The amendment becomes effective as of the date of issuance.

This amendment (1) authorizes changes to the plant Technical Specifications by revising the Minimum Critical Power Ratio Safety Limit to apply to new fuel types, (2) modifies the Rod Drop Accident Design Limit from 1.3%Δ maximum rod worth to 280 calories/gram peak fuel enthalpy rise, (3) approves the use of new Maximum Average Planar Linear Heat Generation Rate curves reflecting 8x8 retrofit fuel reload characteristics and extension of burnup to 40,000 megawatt days per short ton, (4) replaces the Limiting Total Peaking Factor with the Maximum Fraction of Limiting Power Density for adjustment of the APRM flux scram and rod block trip settings, and (5) changes the Reactor Protection System Delay Time from 100 to 50 milliseconds for consistency with the licensing analysis. All other changes correct typographical errors and clarify the basis.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR Section 51.5(d)(4) an environmental impact statement, negative declaration, and environmental impact appraisal need not be prepared in connection with issuance of the amendment.

For further details with respect to this action, see (1) the application for amendment dated September 2, 1980, as supplemented October 3, 1980, (2) Amendment No. 61 to License No. DPR-29, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D. C., and at the Moline Public Library, 504 - 17th Street, Moline, Illinois, for Quad Cities Unit No. 1. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 5th day of December, 1980.

FOR THE NUCLEAR REGULATORY COMMISSION

T. A. Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #2
Division of Licensing