2/23/79

Docket No. 50-254

DISTRIBUTION:
Docket
ORB#3 Rdg
NRR Rdg
NRC PDR
Local PDR
VStello
DEisenhut
BGrimes
HDenton
RVollmer
SSheppard
RBevan
OELD
OI&E (3)
BJones (4)
BScharf (15)
STSG
ACRS (16)
DRoss
CMiles
RDiggs
TERA
JRBuchanan
File
Xtra Copies

Mr. C. Reed
Assistant Vice President
Commonwealth Edison Company
P. O. Box 767
Chicago, Illinois 60690

Dear Mr. Reed:

The Commission has issued the enclosed Amendment No. 50 to Facility Operating License No. DPR-29 for Unit No. 1 of the Quad Cities Nuclear Power Station. This amendment is in response to your request dated November 20, 1978, as supplemented December 15, 1978, and February 14, 1979.

This amendment (1) authorizes operation using 192 assemblies of replacement 8x8R fuel, (2) incorporates revised MCPR limits in response to the plant specific analysis for Reload 4 and (3) modifies License Condition 3.C to revise the end-of-cycle coastdown limits that are appropriate to the analyzed conditions for core Reload 4.

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

Sincerely,

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Enclosures:
1. Amendment No. 50 to
   License No. DPR-29
2. Safety Evaluation
3. Notice

cc w/enclosure:
see next page

7903220227

*SEE ATTACHED FOR CONCURRENCE

OFFICE	ORB#3	ORB#3	OELD	ORB#3		
SURNAME	SSheppard	RBevan:acr	*RGoddard	TIppolito	BGrimes	
DATE	2/23/79	2/23/79	2/23/79	2/23/79	2/23/79	

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

Docket No. 50-254

Mr. C. Reed
Assistant Vice President
Commonwealth Edison Company
P. O. Box 767
Chicago, Illinois 60690

Dear Mr. Reed:

The Commission has issued the enclosed Amendment No. to Facility Operating License No. DPR-29 for Unit No. 1 of the Quad Cities Nuclear Power Station. This amendment is in response to your request dated November 20, 1978, as supplemented December 15, 1978, and February 14, 1979.

This amendment (1) authorizes operation ~~with additional 8 x 8 fuel assemblies,~~ using 192 assemblies of replacement 8x8R fuel, (2) incorporates revised MCPR limits in response to the plant specific analysis for Reload 4 and (3) modifies License Condition 3.C to revise the end-of-cycle coastdown limits that are appropriate to the analyzed conditions for core Reload ~~No.~~ 4.

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

Sincerely,

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

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

cc w/enclosure:
see next page

Concur
R. [illegible]
23.2.79

Mr. Cordell Reed

cc w/enclosures:
Mr. D. R. Stichnoth
President
Iowa-Illinois Gas and
Electric Company
206 East Second Avenue
Davenport, Iowa 52801

Mr. John W. Rowe
Isham, Lincoln & Beale
Counselors at Law
One First National Plaza, 42nd Floor
Chicago, Illinois 60603

Mr. Nick Kalivianakas
Plant Superintendent
Quad-Cities Nuclear Power Station
22710 - 206th Avenue - North
Cordova, Illinois 61242

Anthony Z. Roisman
Natural Resources Defense Council
917 15th Street, N. W.
Washington, D. C. 20005

Moline Public Library
504 - 17th Street
Moline, Illinois 61265

Illinois Department of Public Health
ATTN: Chief, Division of Nuclear
Safety
535 West Jefferson
Springfield, Illinois 62761

Mr. Marcel DeJaegher, Chairman
Rock Island County Board
of Supervisors
Rock Island County Court House
Rock Island, Illinois 61201

Director, Technical Assessment Division
Office of Radiation Programs (AW 459)
US EPA
Crystal Mall #2
Arlington, Virginia 20460

U. S. Environmental Protection Agency
Federal Activities Branch
Region V Office
ATTN: EIS COORDINATOR
230 South Dearborn Street
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 UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 50
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 November 20, 1978, as supplemented December 15, 1978, and February 14, 1979, 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 paragraphs 3.B and 3.C of Facility License No. DPR-29 are hereby amended to read as follows:

   3.B. Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 50, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

   3.C. Restrictions

   Reactor power level shall be limited to maintain pressure margin to the safety valve set points during the worst case pressurization transient. The magnitude of the power limitation, if any, and the point in the cycle at which it shall be applied is specified in the Reload No. 4 licensing submittal for Quad Cities Unit No. 1 (NEDO 24145). Plant operation shall be limited to the operating plan described therein. Subsequent operation in the coastdown mode is permitted to 70% power.

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

FOR THE NUCLEAR REGULATORY COMMISSION

T A Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Attachment:
Changes to the Technical Specifications

Date of Issuance: February 23, 1979

ATTACHMENT TO LICENSE AMENDMENT NO. 50

FACILITY OPERATING LICENSE NO. DPR-29

DOCKET NO. 50-254

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by the captioned Amendment number and contain vertical lines indicating the area of change.

Remove	Insert
1.1/2.1-2	1.1/2.1-2
1.1/2.1-4	1.1/2.1-4
1.1/2.1-5	1.1/2.1-5
1.1/2.1-8	1.1/2.1-8
1.1/2.1-9	1.1/2.1-9
1.2/2.2-1	1.2/2.2-1
3.1/4.1-11	3.1/4.1-11
3.2/4.2-5	3.2/4.2-5
-	3.2/4.2-5a
3.2/4.2-6	3.2/4.2-6
3.2/4.2-7	3.2/4.2-7
3.2/4.2-8	3.2/4.2-8
3.2/4.2-11	3.2/4.2-11
3.2/4.2-12	3.2/4.2-12
3.2/4.2-14	3.2/4.2-14
3.3/4.3-3	3.3/4.3-3
3.3/4.3-4	3.3/4.3-4
3.3/4.3-9	3.3/4.3-9
3.3/4.3-10	3.3/4.3-10
3.5/4.5-10	3.5/4.5-10
3.5/4.5-11	3.5/4.5-11
3.5/4.5-12	3.5/4.5-12
3.5/4.5-14	3.5/4.5-14
3.5/4.5-15	3.5/4.5-15

Add page 3.2/4.2-5a.

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).

curve in Figure 2.1-2, at which point the actual peaking factor value shall be used.

LTPF = 3.06 (7 x 7 fuel assemblies)
3.03 (8 x 8 fuel assemblies)
3.00 (8x8R fuel assemblies)

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:

$$S \leq (.65W + 43)(LTPF/TPF)$$

The definitions used above for the APRM scram trip apply.

C. Reactor low water level scram setting shall be ≥ 143 inches above the top of the active fuel at normal operating conditions.

D. Reactor low water level ECCS initiation shall be 83 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.

1.1 SAFETY LIMIT BASES

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 1.07MCPR > 1.07represents 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.

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 safety limit (MCPR of 1.07 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, 1.07is 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 = 1.07would 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 13.4 kw/ft for 8 x 8 fuel. This constraint is established by Specifications 2.1.A.1 and 3.5.J. Specification 2.1.A.1 established limiting total peaking factors (LTPF) which constrain LHGR's to the maximum values at 100% power and established procedures for adjusting APRM scram settings which maintain equivalent safety margins when the total peak factor (TPF) exceeds the LTPF. 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. Scram times of each control rod are checked each refueling outage, and at least every 32 weeks, 50% are checked to assure adequate inseration times. 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 = 1.07 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 active fuel is defined to be 360 inches above vessel zero (see Bases 3.2).

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 TPF and reactor core thermal power. The scram setting is adjusted in accordance with the formula in Specification 2.1.A.1, when the maximum total peaking factor is greater than the limiting total peaking factor.

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 setting 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 sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power rise. Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks, and because several rods must be moved to change power by a significant percentage of rated power, the rate of power rise is very slow. Generally, the heat flux is 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 logic channels. The IRM is a 5-decade instrument which covers the range of power level between that covered by the SRM and the APRM. The 5 decades are 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 setting is also ranged up.

The most significant sources of reactivity change during the power increase are due to control rod withdrawal. 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 analysis included starting the accident at various power levels. The most severe case involves an initial condition in which the reactor is just subcritical and the IRM system is not yet on scale.

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 scrammed and peak power limited to 1% of rated power, thus maintaining MCPR above 1.07. Based on the above analysis, the IRM provides protection against local control rod withdrawal errors and continous withdrawal of control rods in sequence and 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 rod withdrawal beyond a given point at constant recirculation flow rate to protect against the condition of an MCPR less than 1.07. 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 the APRM scram trip setting, the APRM rod block trip setting is adjusted downward if the maximum total peaking factor exceeds the limiting total peaking factor, 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 1.07 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 1.07 for this transient. For the load rejection from 100% power, the LHGR increases to only 106.5% of its rated value, which results in only a small decrease in MCPR.

1.2/2.1 REACTOR COOLANT SYSTEM

SAFETY LIMIT

Applicability:

Applies to limits on reactor coolant system pressure.

Objective:

To establish a limit below which the integrity of the reactor coolant system is not threatened due to an overpressure condition.

LIMITING SAFETY SYSTEM SETTING

Applicability:

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

Objective:

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

SPECIFICATIONS

Safety Limit

A. The reactor coolant system pressure shall not exceed 1325 psig at any time when irradiated fuel is present in the reactor vessel.

Limiting Safety System Setting

A. Reactor coolant high-pressure scram shall be ≤1060 psig.

B. Primary system safety valve nominal settings shall be as follows:

1 valve at 1115 psig(1)
2 valves at 1240 psig
2 valves at 1250 psig
4 valves at 1260 psig

(1)Target Rock combination safety/relief valve

The allowable setpoint error for each valve shall be ±1%.

TABLE 3.1-4

NOTES FOR TABLES 3.1-1, 3.1-2, AND 3.1-3

1. There shall be two operable trip systems or one operable and one tripped system for each function.

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

   A. Initiate insertion of operable rods and complete insertion of all operable rods within 4 hours.

   B. Reduce power level to IRM range and place mode switch in the Startup/Hot Standby position within 8 hours.

   C. Reduce turbine load and close main steamline isolation valves within 8 hours.

3. An APRM will be considered inoperable if there are fewer than 2 LPRM inputs per level or there are less than 50% of the normal complement of LPRM's to an APRM.

4. Permissible to bypass, with control rod block for reactor protection system reset in refuel and shutdown positions of the reactor mode switch.

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

6. The design permits closure of any one line without a scram being initiated.

7. Automatically bypassed when reactor pressure is <1060 psig.

8. The +8-inch trip point is the water level as measured by the instrumentation outside the shroud. The water level inside the shroud will decrease as power is increased to 100% in comparison to the level outside the shroud to a maximum of 7 inches. This is due to the pressure drop across the steam dryer. Therefore, at 100% power, an indication of +8-inch water level will actually be +1 inch inside the shroud 1 inch on the water level instrumentation is $\geq 504''$ above vessel zero. (See Bases 3.2).

9. Permissible to bypass when first stage turbine pressure is less than that which corresponds to 45% rated steam flow. (<400 psi)

10. Trips upon actuation of the fast-closure solenoid which trips the turbine control valves.

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

12. Channel shared by the reactor protection and containment isolation system.

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) 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 (NEDO 24146). 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, -39° 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 (reference SAR Sections 6.2.7.1 and 14.2.4.2). The instrumentation also covers the full spectrum of breaks and meets the above criteria (reference SAR Sections 6.2.7.1).

The high-drywell pressure instrumentation is a backup to the water level instrumentation and, in addition to initiating ECCS, it causes isolation of Group 2 isolation valves. For the breaks discussed above, this instrumentation will initiate ECCS operation at about the same time as the low low water level instrumentation; thus the results given above are applicable here also. Group 2 isolation valves include the drywell vent, purge, and sump isolation valves. High-drywell pressure activates only these valves because high drywell pressure could occur as the result of non-safety-related causes such as not purging the drywell air during startup. Total system isolation is not desirable for these conditions, and only the valves in Group 2 are required to close. The low low water level instrumentation initiates protection for the full spectrum of loss-of-coolant accidents and causes a trip of Group 1 primary system isolation valves.

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 approach 1.07 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 core control rods in the normal withdrawal sequence. The trips are set so that MCPR is maintained greater than 1.07.

The APRM rod block function, which is set at 12% of rated power, is functional in the Refuel and Startup/Hot Standby modes. 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., it prevents MCPR from decreasing below 1.07 during control rod withdrawals and 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 has been analyzed, and the results show that with the specified trip settings, rod withdrawal is blocked before the MCPR reaches 1.07, thus allowing adequate margin (Reference 1).

Below 30% power, the worst-case withdrawal of a single control rod results in a MCPR greater than 1.07 without rod block action. Thus it 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 1.07.

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.

References

1. GE Topical Report NEDO-24145, "General Electric Boiling Water Reactor Reload No. 4 Licensing Submittal for Quad-Cities Nuclear Power Station (Unit 1)", Section 6.3.3.2, September, 1978.

TABLE 3.2-1

INSTRUMENTATION THAT INITIATES PRIMARY CONTAINMENT ISOLATION FUNCTIONS

Minimum Number of Operable or Tripped Instrument Channels[1]	Instruments	Trip Level Setting	Action[2]
4	Reactor low water[5]	>144 inches above top of active fuel*	A
4	Reactor low low water	>84 inches above top of active fuel*	A
4	High drywell pressure[5]	≤2 psig[3]	A
16	High flow main steamline[5]	≤120% of rated steam flow	B
16	High temperature main steamline tunnel	≤200 ° F	B
4	High radiation main steamline tunnel[6]	≤7 x normal rated power background	B
4	Low main steam pressure[4]	≥850 psig	B
4	High flow RCIC steamline	≤300% of rated steam flow	C
16	RCIC turbine area high temperature	≤200 ° F	C
4	High flow HPCI steamline	≤300% of rated steam flow	D
16	HPCI area high temperature	≤200 ° F	D

Notes

1. Whenever primary containment integrity is required, there shall be two operable or tripped systems for each function, except for low-pressure main steamline which only need be available in the Run position.
2. Action: If the first column cannot be met for one of the trip systems, that trip system shall be tripped.

   If the first column cannot be met for both trip systems, the appropriate actions listed below shall be taken:

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

   B. Initiate an orderly load reduction and have reactor in Hot Standby within 8 hours.

   C. Close isolation valves in RCIC system.

   D. Close isolation valves in HPCI subsystem.
3. Need not be operable when primary containment integrity is not required.
4. The isolation trip signal is bypassed when the mode switch is in Refuel or Startup/Hot Shutdown.
5. This instrumentation also isolates the control room ventilation system.
6. This signal also automatically closes the mechanical vacuum pump discharge line isolation valves.

*Top of active fuel is defined as 360" above vessel zero for all water levels used in the LOCA analysis (see Bases 3.2).

TABLE 3.2-2

INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

Minimum Number of Operable or Tripped Instrument Channels[1]	Trip Function	Trip Level Setting	Remarks
4	Reactor low low water level	$\geq$84 inches (+ 4 inches/ -0inch) above top of active fuel*	1. In conjunction with low-reactor pressure initiates core spray and LPCI. 2. In conjunction with high-drywell pressure 120-second time delay and low-pressure core cooling interlock initiates auto blowdown. 3. Initiates HPCI and RCIC. 4. Initiates starting of diesel generators.
4[4]	High-drywell pressure[2],[3]	≤2 psig	1. Initiates core spray, LPCI, HPCI, and SGTS. 2. In conjunction with low low water level, 120-second time delay, and low-pressure core cooling interlock initiates auto blowdown. 3. Initiates starting of diesel generators. 4. Initiates isolation of control room ventilation.
2	Reactor low pressure	300 psig≤p≤350 psig	1. Permissive for opening core spray and LPCI admission valves. 2. In conjunction with low low reactor water level initiates core spray and LPCI.
	Containment spray interlock		Prevents inadvertent operation of containment spray during accident conditions.
2[3]	2/3 core height	≥2/3 core height	
4[3]	containment high pressure	0.5 psig≤p≤1.5 psig	
2	Timer auto blowdown	≤120 seconds	In conjunction with low low reactor water level, high-drywell pressure, and low-pressure core cooling interlock initiates auto blow-down.
4	Low-pressure core cooling pump dis-charge pressure	75 psig≤p≤100 psig	Defers APR actuation pending confirmation of low-pressure core cooling system operation.
2	Undervoltage on emergency buses	N/A	1. Initiates starting of diesel generators. 2. Permissive for starting ECCS pumps. 3. Removes nonessential loads from buses.

*Top of active fuel is defined as 360" above vessel zero for all water levels used in the LOCA analysis.

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]	≤0.650W + 43[2]
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]	≤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 center-line
3	IRM detector not in Startup position[8]	≥2 feet below core center-line
2[5] [6]	SRM upscale	≤10^5 counts/sec
2[5]	SRM downscale[9]	≥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), RBM upscale, and RBM downscale need not be operable in the Startup/Hot Standby mode. 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 is the reactor recirculation loop flow in percent. 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.

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 not make the core more than 0.013 Δk supercritical.

   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 1.07 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. an end-of-cycle delayed neutron fraction of 0.005,

c. a beginning-of-life Doppler reactivity feedback,

d. the rod scram insertion rate shown in Specification 3.3.C,

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 will be substantially less than 0.013 Δk. Further, the addition of 0.013 Δk worth of reactivity, as a result of a rod drop and 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. However, the 0.013 Δk limit is applied in order to allow room for future reload changes and ease of verification without repetitive technical specification changes.

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 of more fuel rods with MCPR's less than 1.07. 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 1.07. The limiting power transient is that resulting from a turbine stop valve closure with failure of the turbine bypass system. Analysis of this 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 1.07. Reference 1 shows the control rod scram reactivity used in analyzing the transients. Reference 1 should not be confused with the total control rod worth, 18% Δk, as listed in some amendments to the SAR. The 18% Δk value represents the amount of reactivity available for withdrawal in the cold clean core, whereas the control rod worths shown in Reference 1 represent the amount of reactivity available for insertion (scram) in the hot operating core. The minimum 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, 390 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 270 milliseconds. Approximately 70 milliseconds after neutron flux reaches the trip point, the pilot scram valve solenoid deenergizes. Approximately 200 milliseconds later, control rod motion begins. The time to deenergize the pilot valve scram solenoids is measured during the calibration tests required by Specification 4.1. The 200 milliseconds are 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 scram tested at increasing intervals 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 at increasing time intervals provides reasonable assurance of detection of slow drives before system deterioration beyond the limits of Specification 3.3.C. The program was developed on the basis of the statistical approach outlined below and judgment.

The probability that the mean 90% insertion time of a sample of 25 control rod drives will not exceed 0.25 seconds of the mean of all drives is 0.99 at a risk of 0.01. If the mean time exceeds this range or the mean 90% insertion time is greater than 3.5 seconds, an additional sample of drives will be measured to verify the mean performance.

Since the differences between the expected observed mean insertion time and the limit of Specification 3.3.C greatly exceed the expected range, this sampling technique gives assurance that the limits of Specification 3.3.C will not be exceeded. As further assurance that the limits of Specification 3.3.C will not be exceeded, all operable drives will be scram tested to determine compliance to Specification 3.3.C if the enlarged sample of 50 control rods exceeds 4.25 seconds. The 0.75 second margin to the limit is greater than the maximum expected deviation from the mean and therefore gives assurance that the mean will not exceed the limit of Specification 3.3.C. In addition, 50% of the control rods will be checked every 16 weeks to verify the performance and for correlation with the sampling program.

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.

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.

$$LHGR_{max} \quad LHGR_d \; 1 - (\Delta P/P)_{max}(L/L_T)$$

$LHGR_d$ = design LHGR

where:

= 17.5 kW/ft, 7 x 7 fuel assemblies

= 13.4 kW/ft, 8 x 8
8 x 8R fuel assemblies

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

= .035 initial core fuel

= .029 reload 1, 7 x 7 fuel

= .022 reload, 8 x 8 fuel

= .028 reload 1, mixed oxide fuel
= .000 reload 8 x 8R fuel assemblies

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

1.23 (7 x 7 fuel)
1.29 (8 x 8 fuel)
1.32 (8 x 8 BLTA)

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 NEDO-24146, "Loss-of-Coolant Analysis Report for Dresden Units 2, 3 and Quad-Cities Units 1, 2 Nuclear Power Stations, September 1978 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 additionally 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 ft² up to and including 4.18 ft², 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 1. 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.

Should the loss of one RHR pump occur, a nearly full complement of core and containment cooling equipment is available. Three RHR pumps in conjunction with the core spray subsystem will perform the core cooling function. Because of the availability of the majority of the core-cooling equipment, which will be demonstrated to be operable, a 30-day repair period is justified. If the LPCI mode of the RHR system is not available, at least two RHR pumps must be available to fulfill the containment cooling function. The 7-day repair period is set on this basis.

B. RHR Service Water

The containment cooling mode of the RHR system is provided to remove heat energy from the containment in the event of a loss-of-coolant accident. For the flow specified, the containment long-term pressure is limited to less than 8 psig and is therefore more than ample to provide the required heat-removal capability (reference SAR Section 5.2.3.2).

The containment cooling mode of the RHR system consists of two loops, each containing two RHR service water pumps, one heat exchanger, two RHR pumps, and the associated valves, piping, electrical equipment, and instrumentation. Either set of equipment is capable of performing the containment cooling function. Loss of one RHR service water pump does not seriously jeopardize the containment cooling capability, as any one of the remaining three pumps can satisfy the cooling requirements. Since there is some redundancy left, a 30-day repair period is adequate. Loss of one loop of the containment cooling mode of the RHR system leaves one remaining system to perform the containment cooling function. The operable system is demonstrated to be operable each day when the above condition occurs. Based on the fact that when one loop of the containment cooling mode of the RHR system becomes inoperable, only one system remains, which is tested daily, a 7-day repair period was specified.

C. High-Pressure Coolant Injection

The high-pressure coolant injection subsystem is provided to adequately cool the core for all pipe breaks smaller than those for which the LPCI mode of the RHR system or core spray subsystems can protect the core.

The HPCI meets this requirement without the use of offsite electrical power. For the pipe breaks for which the HPCI is intended to function, the core never uncovers and is continuously cooled, thus no cladding damage occurs (reference SAR Section 6.2.5.3). The repair times for the limiting conditions of operation were set considering the use of the HPCI as part of the isolation cooling system.

D. Automatic Pressure Relief

The relief valves of the automatic pressure relief subsystem are a backup to the HPCI subsystem. They enable the core spray subsystem or LPCI mode of the RHR system to provide protection against the small pipe break in the event of HPCI failure by depressurizing the reactor vessel rapidly enough to actuate the core spray subsystems or LPCI mode of the RHR system. The core spray subsystem and the LPCI mode of the RHR system provide sufficient flow of coolant to limit fuel cladding temperatures less than 2200°F, to assure that core geometry remains intact, to limit the core wide clad metal-water reaction to less than 1%, and to limit the calculated local metal-water reaction to less than 17%.

Loss of 1 of the relief valves affects the pressure relieving capability and, therefore, a 7 day repair period is specified. Loss of more than one relief valve significantly reduces the pressure relief capability, thus a 24-hour repair period is specified based on the HPCI system availability during this period.

E. RCIC

The RCIC system is provided to supply continuous makeup water to the reactor core when the reactor is isolated from the turbine and when the feedwater system is not available. Under these conditions the pumping capacity of the RCIC system is sufficient to maintain the water level above the core without any other water system in operation. If the water level in the reactor vessel decreases to the RCIC initiation level, the system automatically starts. The system may also be manually initiated at any time.

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 specified is based on that presented in Reference 3 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. An irradiation growth factor of 0.25% was used as the basis for determining Δ/P in accordance with References 4 and 5.

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, an MCPR of 1.18, 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 limiting value of MCPR stated in this specification is conservatively assumed to exist prior to the initiation of the transients. 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) 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 submittal 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 as 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. I. M. Jacobs and P. W. Marritt, GE Topical Report APED-5736, 'Guidelines for Determining Safe Test Intervals and Repair Times for Engineered Safeguards,' April 1969.
2. "Loss-of-Coolant-Accident Analysis Report for Dresden Units 2, 3 and Quad-Cities Units 1, 2 Nuclear Power Stations," NEDO-24146, September 1978.
3. GE Topical Report NEDM-10735, 'Fuel Densification Effects on General Electric Boiling Water Reactor Fuel,' Section 3.2.1, Supplement 6, August 1973.
4. J. A. Hinds, GE, Letter to V. A. Moore, USAEC, 'Plant Evaluation with GE GEGAP-III,' December 12, 1973.
5. USAEC Report, 'Supplement 1 to the Technical Report on Densification of General Electric Reactor Fuels,' December 14, 1973.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 50 TO FACILITY LICENSE NO. DPR-29

COMMONWEALTH EDISON COMPANY

QUAD CITIES NUCLEAR POWER STATION UNIT NO. 1

DOCKET NO. 50-254

1.0 Introduction

By letter dated November 20, 1978 (Reference 1) and supplemented by letters dated December 15, 1978 (Reference 2), and February 14, 1979 (Reference 3) Commonwealth Edison Company (the licensee) requested amendment to the Technical Specifications appended to Operating License DPR-29 for Quad-Cities Nuclear Power Station, Unit 1 (QC-1). The proposed changes relate to the fourth refueling of QC-1, which involve the replacement of 192 exposed fuel assemblies with a like number of fresh, two water rod, retrofit 8x8 (8x8R) fuel assemblies. Four of the 192 fresh fuel assemblies will be barrier lead test assemblies which are designed to investigate potential fixes for pellet-cladding interaction fuel failure mechanisms (Section 4.0). In support of this reload application, the licensee has submitted a supplemental reload licensing document (Reference 4) prepared by GE, and proposed Technical Specification changes in Reference 1.

This reload is the first for QC-1 to incorporate General Electric's (GE) 8x8R fuel design on a batch basis. The description of the nuclear and mechanical design of the 8x8R fuel and the exposed fuel designs is contained in GE's generic licensing topical report for BWR reloads (Reference 5). Reference 5 also contains a complete set of references to GE's topical reports which describe GE's BWR reload analysis methods for the nuclear, mechanical, thermal-hydraulic, transient and accident calculations, together with information on the applicability of these methods to cores containing a mixture of different fuel designs. Portions of the plant-specific data, such as operating conditions and design parameters which are used in transient and accident calculations, have also been included in the topical report.

Our safety evaluation (Reference 6) of GE's generic reload licensing topical report concluded that the nuclear and mechanical design of the 8x8R fuel and GE's analytical methods for nuclear, thermal-hydraulic, transient and accident calculations, as applied to cores containing mixtures of 7x7, 8x8, and 8x8R fuel, are acceptable. Our acceptance of the nuclear and mechanical design of the standard 8x8 fuel was expressed in the staff's evaluation (Reference 7) of the information in Reference 8.

As part of our evaluation (Reference 6) of Reference 5 we found the cycle-independent input data for the reload transient and accident analyses to be acceptable. The supplementary cycle-dependent information and input data are provided in Reference 4, which follows the format and content of Appendix A of Reference 5.

As a result of the staff's generic evaluation (Reference 6) of a substantial number of safety considerations related to use of 8x8R fuel in mixed core loadings with 8x8 and 7x7 fuel, only a limited number of additional review items are included in this evaluation. These include the plant and cycle-specific input data and results presented in Reference 4, the LOCA-ECCS analysis results for the reload fuel design, and those items identified in our safety evaluation as requiring special attention during reload reviews.

2.0 Evaluation

2.1 Nuclear Characteristics

For Cycle 5, 192 fresh 8x8R fuel bundles, with a bundle average enrichment of 2.65 wt/% U-235 will be loaded into the core. These will replace a like number of exposed fuel assemblies. The remainder of the 724 fuel assembly reload core will consist of the irradiated 7x7 and 8x8 fuel assemblies exposed during previous cycles.

The information provided in Section 6 of Reference 4 indicates that the fuel temperature and void dependent reactivity response of the reconstituted core is not significantly different from that of previous cycles. Additionally, scram effectiveness, Figures 2a and 2b of Reference 4, is also similar to earlier cycles. The 2.0% $\Delta k/k$ calculated shutdown margin for the reconstituted core meets the Technical Specification core subcriticality requirement in the most reactive operating state with the single most reactive control rod fully withdrawn and all other rods fully inserted. Finally, Reference 4 indicates that a boron concentration of 600 ppm in the moderator has been calculated to make the reactor subcritical by at least 4.5% Δk at 20°C, and xenon free conditions. Therefore, the alternate shutdown requirement of the General Design Criteria can be achieved by the Standby Liquid Control System. We have reviewed these analyses and on the bases as stated above find the results to be acceptable.

2.2 Thermal-Hydraulics

2.2.1 Fuel Cladding Integrity Safety Limit MCPR

As stated in Reference 6, for BWR cores which reload with GE's retrofit 8x8R fuel, the allowable minimum critical power ratio (MCPR), resulting from either core-wide or localized abnormal operational transients, is equal to 1.07. With this MCPR safety limit, at least 99.9% of the fuel rods in the core are expected to avoid boiling transition during these transients.

The 1.07 safety limit minimum critical power ratio (SLMCPR) proposed by the licensee represents a .01 increase from the previous 1.06 SLMCPR. The basis for the revised safety limit is addressed in Reference 5. This change continues to meet the recommendations of Standard Review Plan 4.4 and on that basis has been found acceptable in Reference 6. Modifications to the Technical Specification have been incorporated per this finding.

2.2.2 Operating Limit MCPR

Various transient events will reduce the MCPR from its normal operating value. To assure that the fuel cladding integrity safety limit MCPR will not be violated during any abnormal operational transient, the most limiting transients have been reanalyzed by the licensee to determine which event results in the largest reduction in critical power ratio. Each of the events has been conservatively analyzed for fuel types and for the full range of exposure through the cycle.

The calculational methods, which include cycle-independent initial conditions and transient input parameters, are described in Reference 5. Our acceptance of the values used and related transient analysis methods appear in Reference 6. Supplemental cycle-dependent initial conditions and transient input parameters used in the analysis appear in the table in Section 6 and 7 of Reference 4. Our evaluation of the methods used to develop these supplementary transient input values have already been addressed and appear in Reference 6. The overall transient methodology, including cycle-independent transient analysis inputs, provides an adequately conservative basis for the determination of transient ΔCPRs. The transient events analyzed were load rejection without bypass, turbine trip without bypass, feedwater controller failure, loss of 100°F feedwater heating and control rod withdrawal error.

Based on our review of the licensee's submittals, the most limiting abnormal operational transient for all fuel types and exposure intervals is the load rejection without bypass. Therefore, the licensee will be required to meet the following operating limit MCPRs:

Fuel Type	Operating Limit MCPR
7x7	1.23
8x8	1.29
8x8R	1.29

Thus, when the reactor is operated in accordance with the above operating limit MCPRs the 1.07 SLMCPR will not be violated in the event of the most severe abnormal operational transient. This is acceptable to the staff per the finding of the previous section. On this basis, operating limit MCPR Technical Specifications have been established.

In the analysis of the rod withdrawal error (RWE), flow biased upscale rod block monitor (RBM) setpoints are established to assure that the safety limit MCPR is satisfied. Therefore, this setpoint is specified in the Technical Specifications. On the basis of the acceptance of RWE analysis methods in Reference 6 we find the calculated ΔCPR and RBM setpoint for the RWE acceptable.

2.3 Accident Analysis

2.3.1 ECCS Appendix K Analysis

The licensee has reevaluated the ECCS performance in Enclosure IV to Reference 1. This reevaluation provides the bases for relaxation of Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limits. The relaxation is primarily due to the effects of drilled lower tie plates in the 8x8R reload fuel. This reevaluation is based on NRC accepted GE ECCS models and input (Reference 9). The lead plant for this reevaluation is Duane Arnold Energy Center (Reference 10). The justification for the use of this lead plant analysis is the same as for Pilgrim (References 11 and 12) which was found acceptable in Reference 13. Upon our request, the licensee has provided further documentation on the use of Duane Arnold as a lead plant. In our review we find that QC-1 has similar ECCS injection logic as Duane Arnold. We have also previously concluded that the use of the Duane Arnold lead plant analysis for plants of similar power density (Reference 13) and size (Reference 14) is acceptable. This conclusion is based on the fact that a BWR 4 with plugged one inch diameter holes in the core support plate and with drilled lower tie plates (Duane Arnold) is hydraulicly similar to a BWR 3, which never had the one inch core support plate hole, with drilled lower tie plates (Quad-Cities 1). Therefore, the use of the Duane Arnold analysis as a lead plant is acceptable.

In the NRC required confirmatory break spectrum analysis, a longer time period for hot node uncovery was predicted for a 34% of design basis accident (DBA) size break than for the DBA. However, the 34% of DBA size break does not result in higher peak cladding temperature than the DBA because boiling transition and core uncovery for the smaller break occur at a later time after event initiation than for the DBA. Thus, lower decay heat generation is present at the critical time in the calculation. This is the same explanation as the previous QC-1 ECCS performance analysis (Reference 15) which has previously been accepted (Reference 16). The licensee has documented that confirmatory analysis has shown that the DBA is limiting (Reference 3). Therefore, based on the referenced lead plant applicability and the break spectrum verification that the limiting break is the DBA, we find that all requirements of 10 CFR 50.46 and Appendix K to 10 CFR 50.46 will be met when the reactor is operated in accordance with MAPLHGR versus average planar exposure values of Tables 4A through 4H in Enclosure IV to Reference 1 which have been incorporated in the revised Technical Specifications.

2.3.2 Control Rod Drop Accident

The analysis of the control rod drop accident (CRDA) has been performed on a generic (bounding analysis) basis. In our safety evaluation (Reference 6) of GE's generic reload methods (Reference 5) we concluded that the bounding analysis basis is acceptable with the provision that the key input parameter for a plant specific reload are conservatively bound by the analysis assumptions. In the plant specific reload application (Reference 4) the licensee has shown that the maximum incremental control rod worth is conservatively represented in the bounding analysis. This is acceptable on the previously mentioned basis.

2.3.3 Fuel Loading Error

The licensee has also considered the effect of a possible fuel loading error on bundle CPR. An analysis of the most severe misoriented fuel loading error using GE's new methodology (References 17 and 18), which as modified, has been approved (Reference 19) by the staff, shows that the worst possible rotation of a fuel bundle will not cause a violation of the 1.07 safety limit MCPR. Additionally, an analysis of the most severe mislocated fuel bundle with GE's new, approved methodology shows that the worst potential mislocation will not violate the MCPR safety limit. We find the results of these analysis acceptable.

2.4 Overpressure Analysis

The overpressure analysis for the MSIV closure with high flux scram, which is the limiting overpressure event, has been performed in accordance with the requirements of Reference 6. As specified in Reference 6, the sensitivity of peak vessel pressure to failure of one safety valve has also been evaluated. We agree that there is sufficient margin between the peak calculated vessel pressure and the design limit pressure to allow for the failure of at least one valve. In the analysis the licensee has assumed relief valve setpoints with conservative bias which accounts for measurement uncertainty as specified in the revised Technical Specifications. (The only change from previous analyses is a reduction of 10 psig in one safety/relief valve setpoint.) Therefore, the limiting overpressure event as analyzed by the licensee is considered acceptable on the bases outlined in Reference 6.

2.5 Thermal-Hydraulic Stability

A thermal-hydraulic stability analysis was performed with the methods described in Reference 5. The results show that the channel hydrodynamic and reactor core decay ratios at the least stable operating state (corresponding to the intersection of the natural circulation curve and 105% rod line on the power-flow map) are below the 1.0 Ultimate Performance Limit decay ratio proposed by GE.

The staff has expressed generic concerns regarding reactor core thermal-hydraulic stability at the least stable reactor condition. This condition could be reached during an operational transient from high power if the plant were to sustain a trip of both recirculation pumps without a reactor trip. The concerns are motivated by increasing decay ratios as equilibrium fuel cycles are approached and as reload fuel designs change. The staff concerns relate to both the consequences of operating at a decay ratio of 1.0 and the capability of the analytical methods to accurately predict decay ratios.

The General Electric Company is addressing these staff concerns through meetings, topical reports and a stability test program. Although a final test report has not as yet been received by the staff for review, it is expected that the test results will aid considerably in resolving the staff concerns.

For the previous operating cycle, the staff, as an interim measure, added a requirement to the Technical Specifications which restricted planned operation in the natural circulation mode. Continuation of this restriction will also provide a significant increase in the reactor core stability operating margins for the current cycle so that the decay ratio is <1.0 in all operating modes. On the basis of the foregoing, the staff considers the plant thermal-hydraulic stability characteristics to be acceptable.

3.0 Physics Startup Testing

The licensee will perform a series of physics startup tests and procedures to provide assurance that the conditions assumed for the transient and accident analysis calculations will be met. The tests will check that the core is loaded as intended, that the incore monitoring system is functioning as expected, and that the process computer has been reprogrammed to properly reflect changes associated with the reload. The test program is consistent with that previously found acceptable.

4.0 Barrier Lead Test Assemblies

Four of these 192 fresh fuel assemblies are barrier lead test assemblies (BLTA) which are designed to investigate potential fixes for pellet-cladding interaction (PCI) fuel rod failure mechanism. Detailed descriptions and analyses of the BLTAs are given in Enclosure III to Reference 1. The BLTAs have the same 8x8 lattice configuration as the 8x8R fuel assemblies. They differ in that the BLTA's fuel rods consist of two segmented rods, their inside fuel cladding surface is lined with zirconium or copper, and their fuel rods are prepressurized to three atmospheres. The BLTA have been evaluated with specific attention to these differences and the evaluation results show that all design requirements are satisfied. Safety analyses indicate that the BLTAs will have an insignificant effect on core characteristics. On these bases, we find the use of the BLTAs to be acceptable.

5.0 EOC Power Coastdown

In Reference 1, the licensee has proposed EOC power coastdown operation which is justified on the basis of our evaluation (Reference 6) of GE's reload topical (Reference 5). In our evaluation, we did not specifically consider EOC power coastdown operation. We, therefore, do not consider the subject to have completed a generic review and cannot find operation in this mode acceptable on the referenced basis.

In response to our request for additional information (Reference 3), the licensee has referenced previous coastdown mode analysis (References 20 and 21) and has presented an argument of the acceptability of coastdown operations. The referenced analyses are for specific reactor cycles and are, therefore, not directly applicable to this core. The analyses show that the safety margins increase for CPR and overpressurization. These increased safety margins are due to the dominant effect of decreasing total power level during coastdown. The analyses assume a linear power decrease with exposure. This assumption is conservative because actual reactor power will decrease exponentially. In the referenced analyses, the void coefficient becomes less negative during coastdown operation and the scram reactivity becomes less effective as a shutdown mechanism. The impact on ΔCPR is a decrease for the former and an increase for the later change. The referenced analyses show that the overall effect is, as previously stated, increased pressure and thermal safety margin (CPR).

As previously stated, the referenced analyses are not specifically applicable to this plant and cycle. However, we do agree with the licensees argument that the overall trend will be the same. This agreement is restricted to a terminal power level of about 70%. We are confident that at 70% power the scram reactivity insertion will not be degraded sufficiently to result in a transient more severe than that at EOC. For lower power coastdown operations we have requested cycle specific transient analyses or appropriate justification. Currently, the licensee has indicated that they plan to submit the requested analyses or information and our review of operation at powers lower than 70% of rated is pending on this submittal. On the above bases, we find the coastdown operation as restricted in the license condition to be acceptable.

6.0 Linear Heat Generation Rate, 8x8R Fuel

Linear heat generation rate (LHGR) Technical Specifications changes for 8x8R fuel have been made in accordance with the proposals of Reference 5. The design LHGR and maximum power spiking penalty have been previously reviewed and found acceptable in Reference 6. On this basis we find the changes acceptable.

In order to assure compliance with LHGR design limits for the rod withdrawal error, Limiting Total Peaking Factors (LTPF) are established for use in the APRM scram trip and rod block trip setpoints. An LTPF of 3.0 has been calculated using the methods outlined in Reference 5. We have considered this method in our generic Reference 6 review and, thereby, have found it acceptable. On this basis, the specification for this LTPF on 8x8R fuel assemblies is also acceptable.

7.0 Conclusions

Based on our evaluation of the reload application and available information, we conclude that it is acceptable for the licensee to proceed with Cycle 4 operation of Quad Cities Unit No. 1 in the manner proposed.

We have reviewed the proposed changes to the Technical Specifications and find them acceptable.

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 §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.

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 change 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 this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: February 23, 1979

References:

1. Commonwealth Edison (CE) letter to Director of Nuclear Reactor Regulation, USNRC, dated November 20, 1978.

2. CE letter (Turbak) to Director of Nuclear Reactor Regulation, USNRC, dated December 15, 1978.

3. CE letter (Turbak) to Director of Nuclear Reactor Regulation, USNRC, dated February 14, 1979.

4. "Supplemental Reload Licensing Submittal for Quad-Cities Nuclear Power Station Unit 1 Reload 4," NEDO-24145, September 1978.

5. "Generic Reload Fuel Application, General Electric Report," NEDE-24011-P-3, dated March 1978.

6. USNRC letter (Eisenhut) to General Electric (Gridley) dated May 12, 1978, transmitting "Safety Evaluation for the General Electric Topical Report, 'Generic Reload Fuel Application,' (NEDE-24011-P)."

7. "Status Report on the Licensing Topical Report, General Electric Boiling Water Reactor Generic Reload Application for 8x8 Fuel," NEDO-20360, Revision 1 and Supplement 1 by the Division of Technical Review, Office of Nuclear Reactor Regulation, USNRC, April 1975.

8. "General Electric Boiling Water Reactor Generic Reload Application for 8x8 Fuel," NEDO-20360 Revision 1, Supplement 4, April 1, 1976.

9. "Safety Evaluation for General Electric ECCS Evaluation Model Modifications," letter from USNRC (Goller) to GE (Sherwood), dated April 12, 1977.

10. Iowa Electric letter (Liu) to USNRC (Case) Letter No. IE-77-1453, dated July 29, 1977.

11. "Pilgrim Nuclear Power Station Unit 1 Emergency Core Cooling Systems Reevaluation," August 17, 1977.

12. Additional Information on Reload 3 for Pilgrim Unit 1, August 1, 1977.

13. NRC letter (Davis) to Boston Edison Company (Andognini), dated October 17, 1977.

14. NRC letter (Ippolito) to TVA (Hughes), dated November 18, 1978.

15. "Loss-Of-Coolant Accident Analysis Report for Dresden Units 2 and 3 and Quad-Cities Units 1 and 2 Nuclear Power Stations (Lead Plant)," NEDO-24046 Class I, August 1977.

16. NRC Memorandum from Baer to Goller, "Evaluation of Dresden Unit 2 Reload for Cycle 6 Operation (TACS #7084)," December 2, 1978.

17. GE letter (Engle) to NRC (Eisenhut), "Fuel Assembly Loading Error" dated June 1, 1977.

18. GE letter (Engle) to NRC (Eisenhut) dated November 30, 1977.

19. NRC letter (Eisenhut) to GE (Engle) dated May 8, 1978.

20. R. L. Bolger (CECO) letter to B. C. Rusche (NRC), "Quad-Cities Station Unit 2 Proposed Amendment to Facility License No. DPR-30, Docket No. 50-265," dated June 11, 1976.

21. R. L. Bolger (CECO) letter to E. G. Case (NRC), "Dresden Station Unit 2 Proposed Amendment to Facility Operating License No. DPR-19 to Permit Power Coastdown from 70% Power to 40% Power, NRC Docket No. 50-237," dated June 6, 1977.

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. 50 to Facility Operating License No. DPR-29, issued to Commonwealth Edison Company (acting for itself and on behalf of the Iowa-Illinois Gas and Electric Company), which revised the license and Technical Specifications for operation of the Quad Cities Nuclear Power Station Unit No. 1 (the facility) located in Rock Island County, Illinois. The amendment is effective as of its date of issuance.

This amendment (1) authorizes operation using 192 assemblies of replacement 8x8R fuel, (2) incorporates revised MCPR limits in response to the plant specific analysis for Reload 4 and (3) modifies License Condition 3.C to revise the end-of-cycle coastdown limits that are appropriate to the analyzed conditions for core Reload 4.

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 or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the application for amendment dated November 20, 1978, as supplemented December 15, 1978, and February 14, 1979, (2) Amendment No. 50 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 61265. 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 Operating Reactors.

Dated at Bethesda, Maryland, this 23 day of February 1979.

FOR THE NUCLEAR REGULATORY COMMISSION

TA Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors