Docket

APR 21 1975

Docket Nos. 50-254
50-265

Commonwealth Edison Company
ATTN: Mr. J. S. Abel
Nuclear Licensing Administrator -
Boiling Water Reactors
Post Office Box 767
Chicago, Illinois 60690

Gentlemen:

The Commission has issued the enclosed Amendment Nos. 15 and 11 to Facility License Nos. DPR-29 and DPR-30 for Units 1 and 2 of the Quad Cities Nuclear Power Station. These amendments include Change No. 36 to the Technical Specifications and is in response to Commonwealth Edison's request dated December 13, 1974, as supplemented December 20, 1974, January 27, 1975, March 14 and 27, 1975, and April 9, 1975. These amendments also include your request dated May 15, 1974, as supplemented by filings dated October 22 and December 5, 1974.

These amendments (1) incorporate operating limits in the Technical Specifications based on the new General Electric Thermal Analysis Basis (GETAB), and (2) authorize changes to the APRM flux scram and APRM Rod Block Limits.

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

CP
(1)

You are requested to provide an analysis of the symmetry of the power distribution for the operation with reload 1 fuel of Quad Cities Units 1 and 2. This submittal should include an evaluation of the assumption of the TIP uncertainty used in the GETAB evaluation. It is requested that this analysis be provided within 60 days after Quad Cities Unit 2 resumes operation from the current refueling outage.

Sincerely,

Original signed by
Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Reactor Licensing

Enclosures:
1. Amendment 15 to DPR-29
2. Amendment 11 to DPR-30
3. Safety Evaluation
4. Federal Register Notice

cc: see next page

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dtd. 5/15/74, 10/22/74, 12/5/74, 3/14/75,
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

COMMONWEALTH EDISON COMPANY
AND
IOWA-ILLINOIS GAS AND ELECTRIC COMPANY

DOCKET NO. 50-254

QUAD CITIES NUCLEAR POWER STATION UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 15
License No. DPR-29

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

   A. The applications for amendment by Commonwealth Edison Company (the licensee) dated May 15, 1974, as supplemented October 22 and December 5, 1974, and December 13, 1974, as supplemented December 20, 1974, January 27, 1975, March 14 and 27, 1975, and April 9, 1975, comply 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; and

   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.

2. Accordingly, the license is amended by a change 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 and added (respectively) to read as follows:

B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, as revised by issued changes thereto through Change No. 28.

C. Restrictions

Operation of the reactor is not authorized beyond the point in the fuel cycle at which the reactivity insertion rate during a scram is less than that of Curve B on Figure 1 of "Dresden Station Special Report 29, Supplement B" dated March 29, 1974.

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

FOR THE NUCLEAR REGULATORY COMMISSION

A Giambusso

A. Giambusso, Director
Division of Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Change No. 26 to the
Technical Specifications

Date of Issuance: April 21, 1975

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

COMMONWEALTH EDISON COMPANY
AND
IOWA-ILLINOIS GAS AND ELECTRIC COMPANY

DOCKET NO. 50-265

QUAD CITIES NUCLEAR POWER STATION UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 11
License No. DPR-30

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

   A. The applications for amendment by Commonwealth Edison Company (the licensee) dated May 15, 1974, as supplemented October 22 and December 5, 1974, and December 13, 1974, as supplemented December 20, 1974, January 27, 1975, March 14 and 27, 1975, and April 9, 1975, comply 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; and

   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.

2. Accordingly, the license is amended by a change to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 3.B of Facility License No. DPR-30 is hereby amended to read as follows:

B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, as revised by issued changes thereto through Change No. 28.

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

FOR THE NUCLEAR REGULATORY COMMISSION

A. Giambusso

A. Giambusso, Director
Division of Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Change No. 26 to the
Technical Specifications

Date of Issuance: April 21, 1975

ATTACHMENT TO LICENSE AMENDMENT NOS. 11 AND 15

CHANGE NO. 26 TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NOS. DPR-29 AND DPR-30

DOCKET NOS. 50-254 AND 50-265

The following changes relate to the Quad Cities Technical Specifications. Changed areas on the revised pages are shown by a marginal line.

Remove Pages	Insert Pages
3rd page of Table of Contents	3rd page of Table of Contents
2 and 3	2 and 3
5 through 9	5 through 9
11 through 23	11 through 23
27	27
44	44
64 and 65	64 and 65
75 and 83	75 and 83
105A and 105B	105A, 105B, 105C and 105D
109B (and 109C for Unit 2 Only)	109B and 109C
110	110
	113A

TABLE OF CONTENTS (Cont'd)

1.0 DEFINITIONS

E. Instrument Calibration - An instrument calibration means the adjustment of an instrument signal output so that it corresponds, within acceptable range, and accuracy, to a known value(s) of the parameter which the instrument monitors. Calibration shall encompass the entire instrument including actuation, alarm, or trip. Response time is not part of the routine instrument calibration, but will be checked once per operating cycle.

F. Instrument Functional Test - An instrument functional test means the injection of a simulated signal into the instrument primary sensor to verify the proper instrument response alarm, and/or initiating action.

G. Instrument Check - An instrument check is qualitative determination of acceptable operability by observation of instrument behavior during operation. This determination shall include, where possible, comparison of the instrument with other independent instruments measuring the same variable.

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

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

26

J. Limiting Total Peaking Factor - The Limiting Total Peaking Factor (LTPF) is the lowest Total Peaking Factor which limits a fuel type to a Linear Heat Gneration Rate (LHGR) corresponding to the operating limit at 100% power.

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

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

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

1.0 DEFINITIONS

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

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

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

*

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

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

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

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

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

2. At least one door in each airlock is closed and sealed.

3. All automatic containment isolation valves are operable or deactivated in the isolated position.

4. All blind flanges and manways are closed.

* M not included

may not in itself result in serious consequences but it indicates an operational deficiency subject to regulatory review.

Y. Secondary Containment Integrity - Secondary containment integrity means that the reactor building is intact and the following conditions are met:

1. At least one door in each access opening is closed.

26

2. The standby gas treatment system is operable.

3. All reactor building automatic ventilation system isolation valves are operable or are secured in the isolated position.

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

1. Hot Shutdown means conditions as above with reactor coolant temperature greater than 212°F.

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

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

BB. Total Peaking Factor - The Total Peaking Factor (TPF) is the highest product of radial, axial, and local peaking factors simultaneously operative at any segment of fuel rod.

CC. Transition Boiling - Transition boiling means the boiling regime between nucleate and film boiling. Transition boiling is the regime in which both nucleate and film boiling occur intermittently with neither type being completely stable.

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

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

1.1 SAFETY LIMIT

1.1 FUEL CLADDING INTEGRITY

Applicability

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

Objective

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

26

Specifications

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

The existence of a minimum critical power ratio (MCPR) less than 1.06 shall constitute violation of the fuel cladding integrity safety limit.

2.1 LIMITING SAFETY SYSTEM SETTING

2.1 FUEL CLADDING INTEGRITY

Applicability

The Limiting Safety System Settings apply to trip settings of the instruments and devices which are provided to prevent the fuel cladding integrity Safety Limits from being exceeded.

Objective

The objective of the Limiting Safety System Settings is to define the level of the process variables at which automatic protective action is initiated to prevent the fuel cladding integrity Safety Limits from being exceeded.

26

Specifications

A. Neutron Flux Trip Settings

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

1.1 SAFETY LIMIT	2.1 LIMITING SAFETY SYSTEM SETTING

1. <u>APRM Flux Scram Trip Setting (Run Mode)</u>

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

$$S \leq \left[.65W + 55\right]\left[\frac{LTPF}{TPF}\right]$$

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

where:

S = setting in per cent of rated power

W = per cent of drive flow required to produce a rated core flow of 98 Mlb/hr.

TPF = LTPF unless the combination of power and peak LHGR is above the curve in Figure 2.1-2 at which point the actual peaking factor value shall be used.

LTPF = 3.06 (7X7 fuel assemblies)

3.03 (8X8 fuel assemblies)

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

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.

26

1.1 SAFETY LIMIT

26

B. Core Thermal Power Limit (Reactor Pressure ≤ 800 psig)

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

C. Power Transient

1. The neutron flux shall not exceed the scram setting established in Specification 2.1.A for longer than 1.5 seconds as indicated by the process computer.

2. When the process computer is out of service, this safety limit shall be assumed to be exceeded if the neutron flux exceeds the scram setting established by Specification 2.1.A and a control rod scram does not occur.

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.

2.1 LIMITING SAFETY SYSTEM SETTING

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.

26

B. APRM Rod Block Setting

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

$$S \leq \left[.65W + 43\right] \left[\frac{LTPF}{TPF}\right]$$

The definitions used above for the APRM scram trip apply.

1.1 SAFETY LIMIT	2.1 LIMITING SAFETY SYSTEM SETTING
	C. Reactor low water level scram setting shall be $\geq$ 143" above the top of the active fuel at normal operating conditions.
	D. Reactor low water level ECCS initiation shall be 83" ($^{+4"}_{-0"}$) above the top of the active fuel at normal operating conditions.
	E. Turbine stop valve scram shall be $\leq$ 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 $\leq$ 10% valve closure from full open.
	H. Main steamline low pressure initiation of main steamline isolation valve closure shall be $\geq$ 850 psig.
	I. Turbine EHC control fluid low pressure scram on loss of control oil pressure shall be set at greater than or equal to 900 psig.
	J. Condenser low vacuum scram shall be set at $\geq$ 23 in. Hg Vacuum.

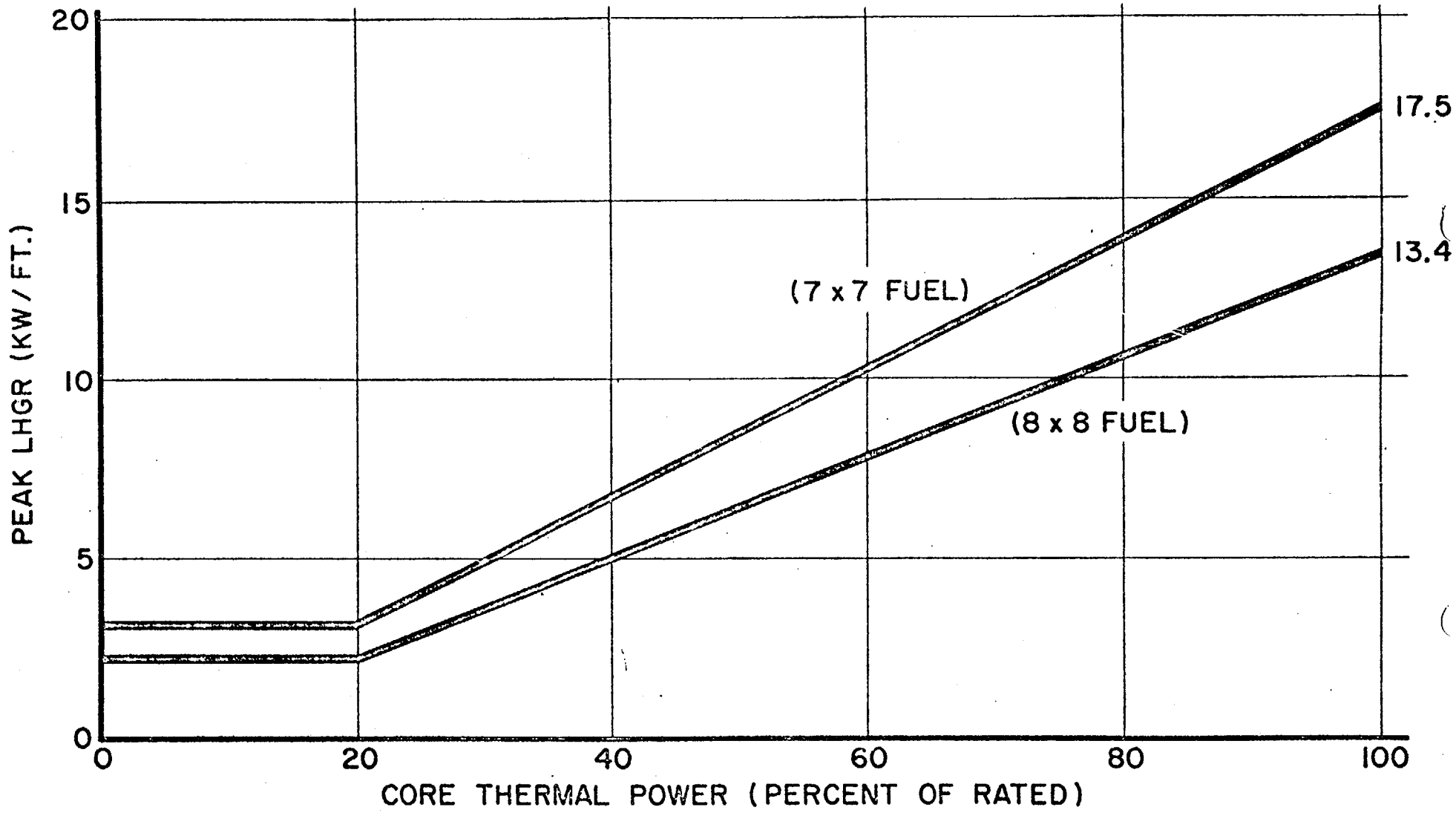

Figure 2.1-2
PEAK LHGR VERSUS CORE THERMAL POWER
FOR A LIMITING TOTAL PEAKING FACTOR

1.1 Safety Limit Bases

FUEL CLADDING INTEGRITY

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 1.06. MCPR > 1.06 represents a conservative margin relative to the conditions required to maintain fuel cladding integrity.

The fuel cladding is one of the physical barriers which separate
26 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.

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

Onset of transition boiling results in a decrease in heat transfer from the clad and, therefore, elevated clad temperature and the possibility of clad 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

Safety Limit Bases

1.1.A Reactor Pressure > 800 psig and Core Flow ≥ 10% of Rated. (cont'd)

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 1.06) has sufficient conservatism to assure that in the event of an abnormal operational transient initiated from a 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, 1.06, is derived from a detailed statistical analysis considering all of the uncertainties in monitoring the core operating state including uncertainty in the boiling transition correlation. See e. g. Reference (1).

Because the boiling transition correlation is based on a large quantity of full scale data there is a very high confidence that operation of a fuel assembly at the condition of MCPR = 1.06 would not produce boiling transition.

However, if boiling transition were to occur, clad 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 clad 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 7x7 fuel and 13.4 fw/ft for 8x8 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 max which cannot be exceeded under steady power operation.

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(1) NEDO-20693. "General Electric Boiling Water Reactor Reload No. 1 Licensing Submittal for Quad Cities Nuclear Power Station Unit 2."

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 28×10^3 lbs/hr. bundle flow, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.56 psi driving head will be greater than 28×10^3 lbs/hr. Full scale ATLAS test data taken at
26 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 8-9 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
26 each refueling outage and at least every 32 weeks 50% are checked to assure adequate insertion 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
26 design limit of MCPR = 1.06 is not exceeded. Thus, use of a 1.5 second limit provides additional margin.

1.1 Safety Limit Bases

1.1.C Power Transient (cont'd)

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 clad perforation. The core will be cooled sufficiently to prevent clad 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.

2.1 Limiting Safety System Setting Bases

FUEL CLADDING INTEGRITY

26 The abnormal operational transients applicable to operation of the units have been analyzed throughout the spectrum of planned operating conditions up to the rated thermal power condition of 2511 MWt. In addition, 2511 MWt is the licensed maximum steady-state power level of the units. This maximum steady-state power level will never knowingly be exceeded.

26 Conservatism is incorporated in the transient analyses in estimating the controlling factors, such as void reactivity coefficient, control rod scram worth, scram delay time, peaking factors, and axial power shapes. These factors are selected conservatively with respect to their effect on the applicable transient results as determined by the current analysis model. This transient model, evolved over many years, has been substantiated in operation as a conservative tool for evaluating reactor dynamic performance. Results obtained from a General Electric boiling water reactor have been compared with predictions made by the model. The comparisons and results are summarized in Reference 2.

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(2) Linford, R. B., "Analytical Methods of Plant Transient Evaluations for the General Electric Boiling Water Reactor," NEDO-10802, Feb., 1973.

26

2.1 Limiting Safety System Setting Bases

Fuel Cladding Integrity (cont'd)

26 The absolute value of the void reactivity coefficient used in the analysis is conservatively estimated to be about 25% greater than the nominal maximum value expected to occur during the core lifetime. The scram worth used has been derated to be equivalent to approximately 80% of the total scram worth of of the control rods. The scram delay time and rate of rod insertion allowed by the analyses are conservatively set equal to the longest delay and slowest insertion rate acceptable by Technical Specifications. The effect of scram worth, scram delay time and rod insertion rate, all conservatively applied, are of greatest significance in the early portion of the negative reactivity insertion. The rapid insertion of negative reactivity is assured by the time requirements for 5% and 25% insertion. By the time the rods are 60% inserted, approximately four dollars of negative reactivity have been inserted which strongly turns the transient, and accomplishes the desired effect. The times for 50% and 90% insertion are given to assure proper completion of the expected performance in the earlier portion of the transient, and to establish the ultimate fully shutdown steady-state condition.

This choice of using conservative values of controlling parameters and initiating transients at the design power level, produces more pessimistic answers than would result by using expected values of control parameters and analyzing at higher power levels.

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

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

For analyses of the thermal consequences of the transients, the MCPR's stated in paragraph 3.5.K are conservatively assumed to exist prior to initiation of the transients.

A. Neutron Flux Trip Settings

1. APRM Flux Scram Trip Setting (Run Mode)

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

2.1.A. Neutron Flux Trip Settings

1. APRM Flux Scram Trip Setting (Run Mode) (cont'd)

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.
26 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
26 (Refuel or Start & Hot Standby Mode)

For operation in the startup mode while the reactor is at low pressure, the APRM scram setting of 15 percent of rated power
26 provides adequate thermal margin between the the setpoint and the safety limit, 25 percent of rated. The margin is adequate to accommodate anticipated maneuvers associated with power plant startup. Effects of increasing pressure at zero or low void content are minor, cold water from sources available during startup is not much colder than that already in the system, temperature coefficients are small, and control rod patterns are constrained to be uniform by operating procedures backed up by the rod worth minimizer. 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 percent of rated power per minute, and the APRM system would be more than adequate to assure a scram before the power could exceed the safety limit. The 15 percent 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.

26 3. IRM Flux Scram Trip Setting

The IRM system consists of 8 chambers, 4 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 of a decade in size.

2.1.A. Neutron Flux Trip Setting

3. IRM Flux Scram Trip Setting (cont'd)

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 a 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 accomodate 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 provided 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.

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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 one percent of rated power, thus maintaining MCPR above 1.06. 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.

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2.1.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 a MCPR less than 1.06. 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 in-core 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.

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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 clad temperature to well below the clad melting temperature to assure that core geometry remains intact and to limit any clad 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.

26 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 percent of valve closure from full open, the resultant increase in surface heat flux is
26 limited such that MCPR remains above 1.06 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

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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 1.06 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.

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

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

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

J. Condenser Low Vacuum Scram - Loss of condenser vacuum occurs when the condenser can no longer handle the heat input. Loss of condenser vacuum initiates a closure of the turbine stop valves and turbine bypass valves which eliminates the heat input to the condenser. Closure of the turbine stop and bypass valves causes a pressure transient, neutron flux rise, and an increase in surface heat flux. To prevent the clad safety limit from being exceeded if this occurs, a reactor scram occurs on turbine stop valve closure in the run mode. The turbine stop valve closure scram function alone is adequate to prevent the clad safety limit from being exceeded in the event of a turbine trip transient with bypass closure. Ref. Section 4.4.3 SAR. The condenser low vacuum scram is anticipatory to the stop valve closure scram and causes a scram before the stop valves are closed and thus the resulting transient is less severe. Scram occurs in the run mode at 23" Hg vacuum, stop valve closure occurs at 20" Hg vacuum and bypass closure at 7" Hg vacuum.

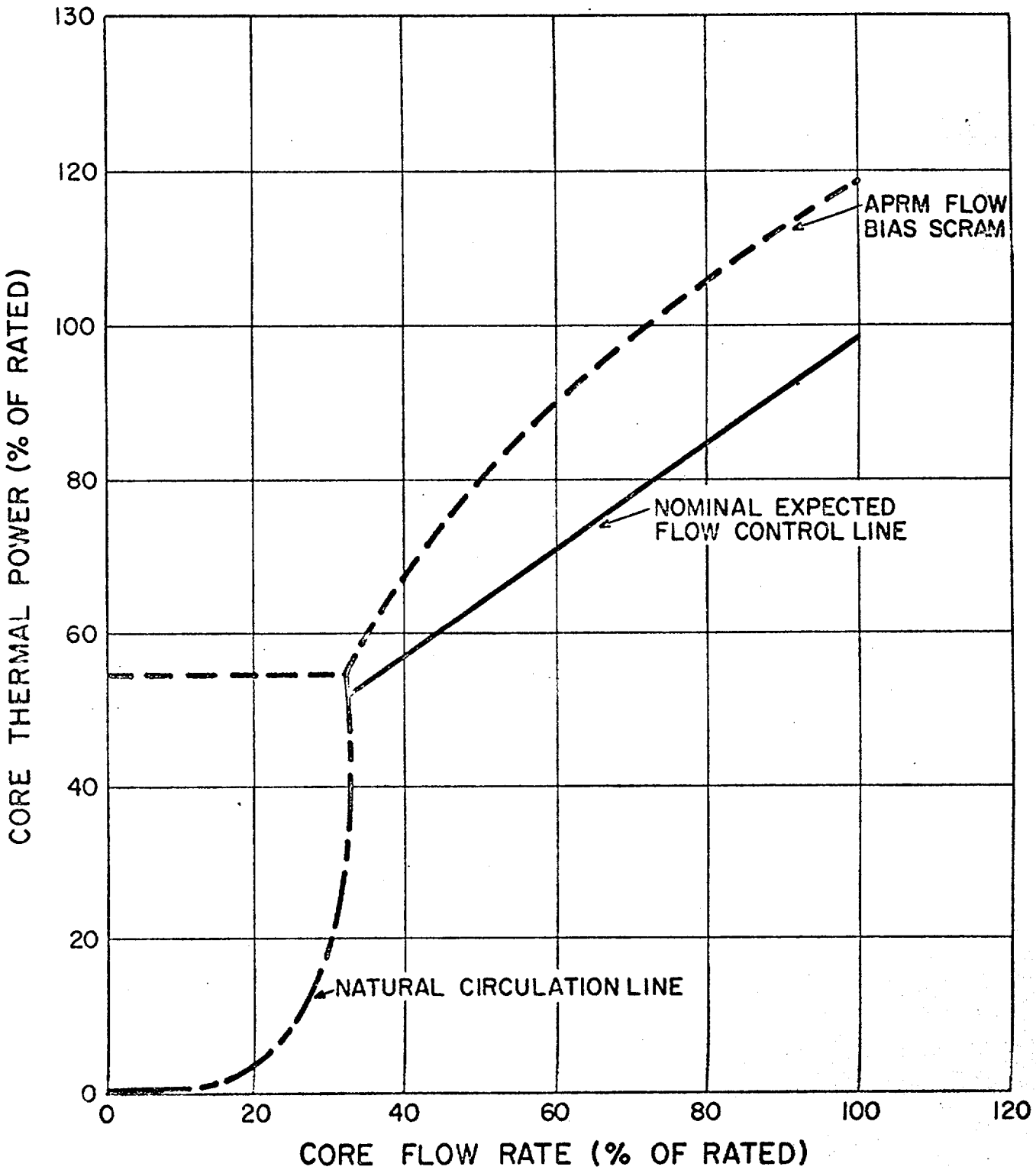

FIGURE 2.1-3 APRM FLOW BIAS SCRAM RELATIONSHIP TO NORMAL OPERATING CONDITIONS

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3.1 LIMITING CONDITION FOR OPERATION	4.1 SURVEILLANCE REQUIREMENT

3.1 REACTOR PROTECTION SYSTEM

Applicability:

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

Objective:

To assure the operability of the reactor protection system.

Specification:

A. The setpoints, minimum number of trip systems, and minimum number of instrument channels that must be operable for each position of the reactor mode switch shall be as given in Table 3.1.1. The system response times from the opening of the sensor contact up to and including the opening of the trip actuator contacts shall not exceed 100 milliseconds.

B. During operation with a Limiting Total Peaking Factor, either:

a. The APRM scram and rod block settings shall be reduced to the values given by the equations in Specifications 2.1.A.1 and 2.1.B; or

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b. The power distribution shall be changed such that a Limiting Total Peaking Factor no longer exists.

4.1 REACTOR PROTECTION SYSTEM

Applicability:

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

Objective:

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

Specification:

A. Instrumentation systems shall be functionally tested and calibrated as indicated in Tables 4.1.1 and 4.1.2, respectively.

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B. Daily during reactor power operation, the core power distribution shall be checked for Limiting Total Peaking Factor (LTPF).

4.1 Surveillance Requirements Bases (cont'd)

Switch in Shutdown, Manual Scram, High Water Level in Scram Discharge Volume, Main Steam-line Isolation Valve Closure, Turbine Control Valve Fast Closure and Turbine Stop Valve Closure. All of the devices or sensors associated with these scram functions are simple on-off switches and, hence, calibration is not applicable; i.e., the switch is either on or off. Based on the above, no calibration is required for these instrument channels.

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

3.2 Limiting Condition for Operations Bases (cont'd)

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" mode 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 clad temperatures are much less than 1500°F; thus, there are no fission products available for release other than those in the reactor water. Ref. 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 and 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. 26

The instrumentation which initiates ECCS action is arranged in a 1 out of 2 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 is met by virtue of the fact that redundant core cooling functions are provided; e.g., two core 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. 26 26

The control rod block functions are provided to prevent excessive control rod withdrawal so that MCPR does not approach 1.06. The trip logic for this function is 1 out of n; e.g., any trip on one of the six APRM's, 8 IRM's, or 4 SRM's will result in a rod block. The minimum instrument channel requirements assure sufficient instrumentation to assure 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. 26

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 1.06. 26

3.2 Limiting Condition for Operations Bases (cont'd)

The APRM rod block function which is set at 12% of rated power is functional in the refuel and Startup/Hot Standby mode. This control rod block provides the same type of protection in the Refuel and Startup/Hot Standby mode as the APRM flow biased rod block does in the run mode; i.e., it prevents MCPR from decreasing below 1.06 during control rod withdrawals and prevents control rod withdrawal before a scram is reached. 26

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 1.06 thus allowing adequate margin, Ref. (1). 26

Below 70 percent power, the worst case withdrawal of a single control rod results in a MCPR greater than 1.06 without rod block action. Thus, below this power level it is not required. 26

The IRM rod block function provides local as well as gross core protection. The scaling arrangement is such that trip setting is less than a factor of 10 above the indicated level. Analysis of the worst case accident results in rod block action before MCPR approaches 1.06. 26

(1) NEDO-20693, "General Electric Boiling Water Reactor Reload No. 1 Licensing Submittal for Quad Cities Nuclear Power Station (Unit 2)" December 1974. Section 6.3.3.2. 26

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

The SRM rod block with $\leq$ 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. Ref. 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.

3.3 LIMITING CONDITION FOR OPERATION

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

   a. Both RBM channels shall be operable; or

   b. Control rod withdrawal shall be blocked; or

   c. The operating power level shall be limited so the the MCPR will remain above 1.06 assuming a single error that results in complete withdrawal of any single operable control rod.

26

C. Scram Insertion Times

1. The average scram insertion time, based on the de-energization 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	Avg. Scram Insertion Times (sec)
5	0.375
20	0.900
50	2.00
90	3.50

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4.3 SURVEILLANCE REQUIREMENT

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. 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.

5. The Rod Block Monitor (RBM) is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power operation. Two channels are provided, and one of these may be bypassed from the console for maintenance and/or testing. Tripping of one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage. This system backs up the operator who withdraws control rods according to a written sequence. The specified restrictions with one channel out of service conservatively assure that fuel damage will not occur due to rod withdrawal errors when this condition exists. During reactor operation with certain limiting control rod patterns, the withdrawal of a designated single control rod could result in one or more fuel rods with MCPRs less than 1.06. During use of such patterns, it is judged that testing of the RBM system prior to withdrawal of such rods to assure its operability 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.

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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 1.06. 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 1.06. 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

(1) "Dresden Station Special Report No. 29, Supplement B", Figure 1.

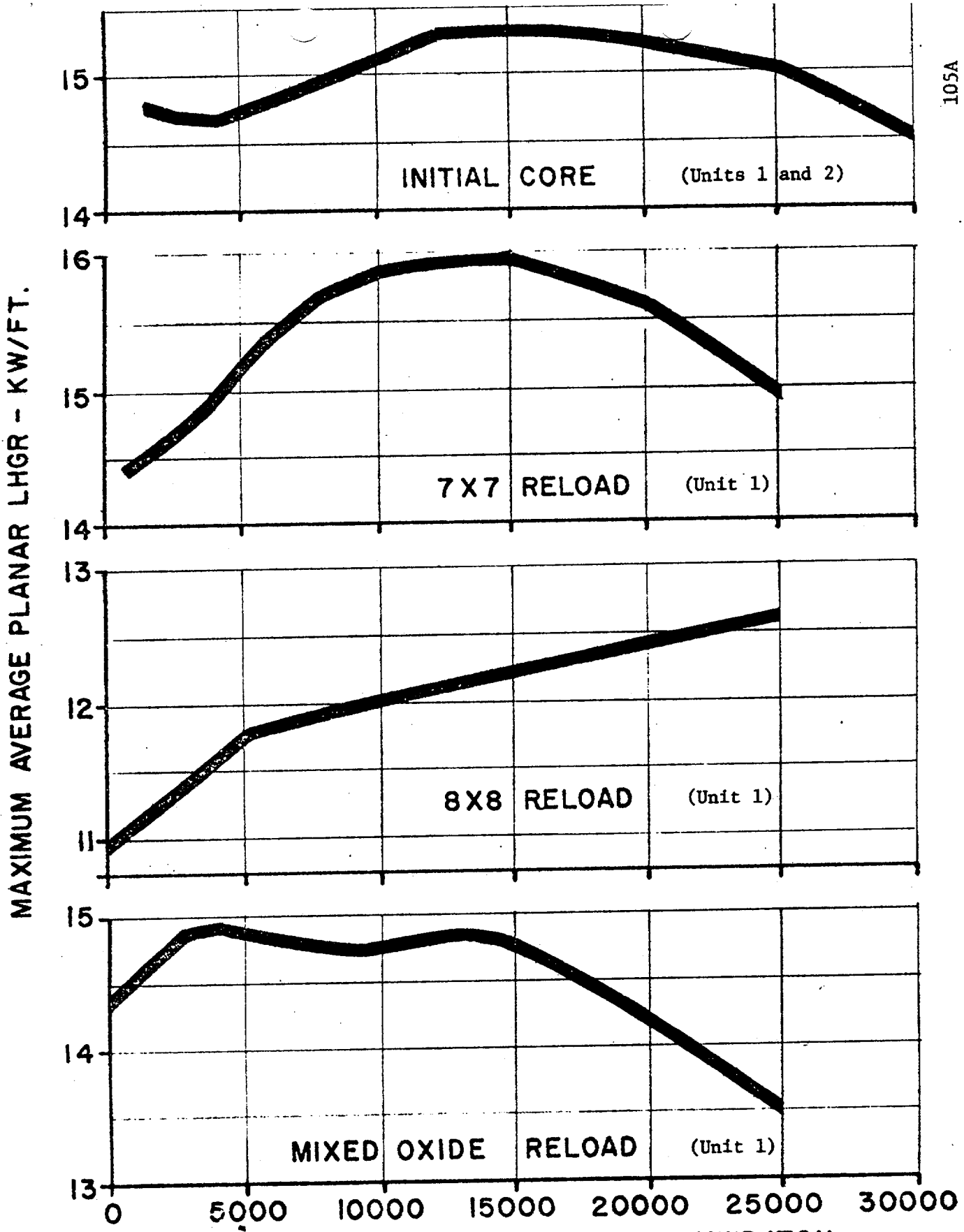

FIGURE 3.5.1 MAXIMUM ALLOWABLE PLANAR LHGR APPLICABLE TO QUAD-CITIES INITIAL AND RELOAD FUEL

3.5 LIMITING CONDITIONS FOR OPERATION

J. Local LHGR

During steady state power operation, the linear heat generation rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed the maximum allowable LHGR as calculated by the following equation.

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

$LHGR_d$ = Design LHGR

= 17.5 kw/ft, 7X7 fuel assemblies

= 13.4 kw/ft, 8X8 fuel assemblies

$\left(\frac{\Delta P}{P}\right)_{max}$ = Maximum power spiking penalty

= .035 initial core fuel

= .029 reload 1, 7X7 fuel

26 = .022 reload, 8X8 fuel

= .028 reload 1, mixed oxide fuel

L_T = Total Core Length

= 12 ft

L = Axial distance from bottom of core

4.5 SURVEILLANCE REQUIREMENTS

J. Local LHGR

26 Daily during steady state power operation above 25 per cent of rated thermal power, the Local LHGR shall be checked.

3.5 LIMITING CONDITION FOR OPERATION

K. Minimum Critical Power Ratio (MCPR)

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

1.29 (7X7 fuel)
1.35 (8X8 fuel)

at rated power and flow. 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. 26

4.5 SURVEILLANCE REQUIREMENTS

K. Minimum Critical Power Ratio (MCPR)

The MCPR shall be determined daily during steady state power operation above 25% of rated thermal power. 26

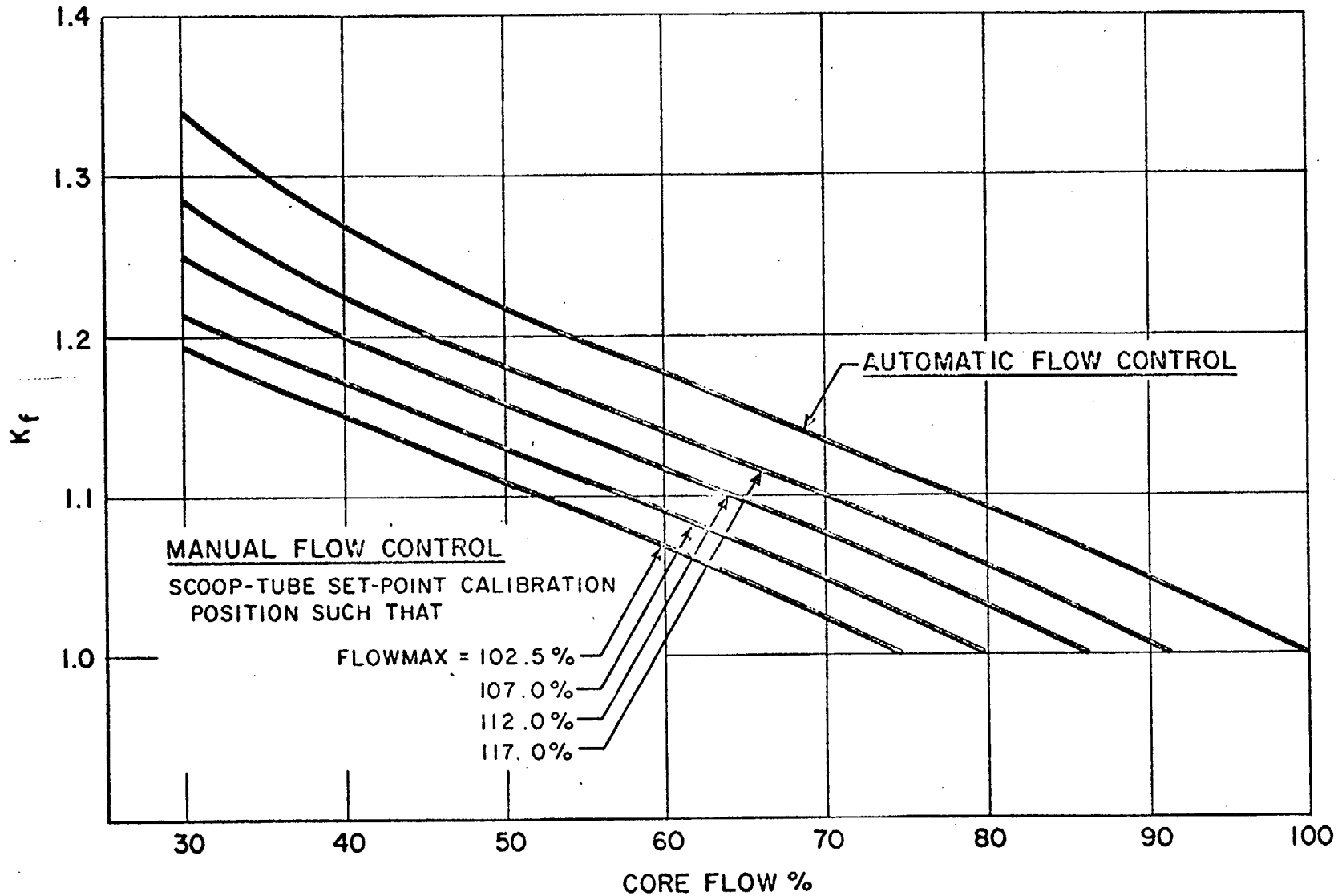

FIGURE 3.5-2 K_f FACTOR

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 2300°F limit specified in the Interim Acceptance Criteria (IAC) issued in June 1971 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 dependent secondarily on the rod to rod power distribution within an assembly. Since expected local variations in power distribution within a fuel assembly affect the calculated peak clad temperature by less than ± 20°F relative to the peak temperature for a typical fuel design, the limit on the average planar LHGR is sufficient to assure that calculated temperatures are below the IAC limit.

The maximum average planar LHGRs shown in Figure 3.5.1 are based on calculations employing the models described in Reference 1 as modified by Reference 2, and authorized in Reference 3.

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 4, 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 $\Delta P/P$ in accordance with References 5 and 6.

(1) NEDM-10735, "Fuel Densification Effects on General Electric Boiling Water Reactor Fuel," Aug. 1973.
(2) NEDO-20181, "GEGAP-III: A Model for the Prediction of Pellet-Cladding Thermal Conductance in BWR Fuel Rods," Nov. 1973.
(3) D.J. Skovholt (USAEC) Letter to J.S. Abel (CE Co.) Dec. 5, 1973.
(4) NEDM-10735, "Fuel Densification Effects on General Electric Boiling Water Reactor Fuel," Section 3.2.1, Supplement 6, Aug. 1973.
(5) J.A. Hinds (GE) letter to V.A. Moore (USAEC), "Plant Evaluation with GE GEGAP-III," Dec. 12, 1973.
(6) USAEC Report, "Supplement 1 to the Technical Report on Densification of General Electric Reactor Fuels," Dec. 14, 1973.

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, a 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 limiting transient which determines the required steady state MCPR limits is the turbine trip event assuming failure of the turbine bypass valves with a scram initiated by the turbine stop valve position switches.

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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.

4.5 Surveillance Requirements Bases:

The testing interval for the core and containment cooling systems is based on a quantitative reliability analysis, judgment, and practicality. The core cooling systems have not been designed to be fully testable during operation. For example, the core spray final admission valves do not open until reactor pressure has fallen to 350 psig. Thus, during operation, even if high drywell pressure were simulated, the final valves would not open. In the case of the HPCI, automatic initiation during power operation would result in pumping cold water into the reactor vessel which is not desirable.

The systems can be automatically actuated during a refueling outage and this will be done. To increase the availability of the individual components of the core and containment cooling systems, the components which make up the system, i.e., instrumentation, pumps, valve operators, etc., are tested more frequently. The instrumentation is functionally tested each month. Likewise the pumps and motor-operated valves are also tested each month to assure their operability. The combination of a yearly simulated automatic actuation test and monthly tests of the pumps and valve operators is deemed to be adequate testing of these systems.

With components or subsystems out-of-service overall core and containment cooling reliability is maintained by demonstrating the operability of the remaining cooling equipment. The degree of operability to be demonstrated depends on the nature of the reason for the out-of-service equipment. For routine out-of-service periods caused by preventative maintenance, etc., the pump and valve operability checks will be performed to demonstrate operability of the remaining components. However, if a failure, design deficiency, etc., caused the out-of-service period, then the demonstration of operability should be thorough enough to assure that a similar problem does not exist on the remaining components. For example, if an out-of-service period were caused by failure of a pump to deliver rated capacity due to a design deficiency, the other pumps of this type might be subjected to a flow rate test in addition to the operability checks.

The surveillance requirements to ensure the discharge piping of the core spray, LPCI mode of the RHR, HPCI, and RCIC systems are filled provides for a visual observation that water flows from a high point vent. This ensures that the line is in a full condition. Between the monthly intervals at which the lines are vented, instrumentation has been provided to monitor

I. Average Planar LHGR

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

J. Local LHGR

The LHGR as a function of core height shall be checked daily during reactor operation at greater than or equal to 25 per cent power to determine if fuel burnup or control rod movement has caused changes in power distribution. For LHGR to be a limiting value below 25 per cent rated thermal power, the MTPF would have to be greater than 10 which is precluded by a considerable margin when employing any permissible control rod pattern.

K. Minimum Critical Power Ratio (MCPR)

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

The daily requirement for calculating MCPR above 25 per cent rated thermal power is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes.

In addition, the K_f correction applied to the LCO provides margin for flow increases from low flows.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NOS. 15 AND 11 TO FACILITY OPERATING LICENSE NOS. DPR-29 AND DPR-30

(CHANGE NO. 26 TO THE TECHNICAL SPECIFICATIONS)

COMMONWEALTH EDISON COMPANY AND

IOWA-ILLINOIS GAS AND ELECTRIC COMPANY

QUAD CITIES STATION UNITS 1 AND 2

DOCKET NOS. 50-254 AND 50-265

INTRODUCTION

By application dated December 13, 1974, Commonwealth Edison (CE) requested that the licenses for Quad Cities Units 1 and 2 be amended to include operating limits based on the new General Electric Thermal Analysis Basis (GETAB) described in General Electric Company report NEDO-10958[1]. Analyses of the effect of applying GETAB to normal operation, anticipated transients and accidents were attached to the CE letter dated December 20, 1974, in support of operation of Quad Cities Unit 2 with 8 x 8 reload fuel. Proposed changes to the Technical Specifications of Facility Operating Licenses DPR-29 and DPR-30 for Quad Cities Units 1 and 2 based on GETAB were submitted by a letter dated January 27, 1975. Supplemental information related to GETAB was submitted by letters dated March 14, 1975, March 27, 1975 and April 9, 1975.

The proposed changes involve the adoption of a new transition boiling correlation termed GEXL which would replace the Hench-Levy critical heat flux correlation as the basis for determining the thermal-hydraulic conditions which would result in a departure from nucleate boiling. Notice of the proposed issuance of the amendments was issued on February 3, 1975, and published in the Federal Register on February 10, 1975 (40 F.R. 6240).

(1) "General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Application, " NEDO-10958, November, 1973.

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In addition to evaluation of the use of GETAB, this evaluation also considers proposed changes to the Technical Specifications of License Nos. DPR-29 and DPR-30 requested by application dated May 15, 1974, as supplemented October 22 and December 5, 1974, to modify limitations related to the average power range monitor (APRM) flux scram, and the APRM rod block and to change the definitions for limiting total peaking factor.

EVALUATION

1. General Electric Thermal Analysis Basis (GETAB)

One of the safety requirements for light water cooled nuclear reactors is prevention of damage to the fuel cladding. To prevent damage to the fuel cladding, light water cooled reactors must be designed and operated such that during normal operation and anticipated transients the heat transfer rate from the fuel cladding to the coolant are sufficient to prevent overheating of the fuel cladding. Although transition boiling would not necessarily result in damage to boiling water reactor (BWR) fuel rods, historically it has been used as a fuel damage limit because of the large reduction in heat transfer rate when film boiling occurs. A critical power ratio (CPR) is defined which is the ratio of that assembly power which causes some point in the assembly to experience transition boiling to the assembly power at the reactor condition of interest. The minimum critical power ratio (MCPR) is the critical power ratio corresponding to the most limiting fuel assembly in the core. The fuel assembly power at which boiling transition would be predicted to occur, using the GEXL correlation, is termed the critical power. The GEXL transition boiling correlation is more recent than the previously used Hensch-Levy critical heat flux correlation and is based on an extensive data base. The methods for applying the GEXL correlation to determine thermal limits has been termed the General Electric Thermal Analysis Basis (GETAB). We have accepted the GEXL correlation and the GETAB methods in a previous report(2) as a basis for establishing the safety limit and limiting conditions for operation related to prevention of fuel damage for General Electric BWR 8 x 8 and 7 x 7 fuel. To apply GETAB to the Technical Specifications involves 1) establishing the fuel damage safety limit, 2) establishing limiting conditions of operation such that the safety limit is not exceeded for normal operation and anticipated transients, and 3) establishing limiting conditions for operation such that the initial conditions assumed in accident analyses are satisfied.

(2) "Review and Evaluation of GETAB (General Electric Thermal Analysis Basis) for BWRs," Division of Technical Review, Directorate of Licensing, United States Atomic Energy Commission, September, 1974.

In applying the GEXL correlation, the safety limit is based on a statistical analysis of the uncertainties in the GEXL correlation and other input parameters such that if an event occurred which caused the limiting fuel bundle to reach the safety limit, boiling transition would not occur in 99.9% of the fuel. To meet these conditions, the MCPR must be greater than the unity by a ratio determined by the magnitude of the uncertainties. For Quad Cities Units 1 and 2, the input list of uncertainty effects of the core operating parameters and calculated parameters associated with the GEXL correlation plus the GETAB relative bundle power histogram used in the statistical analysis were presented in reference 3.

We have reviewed those uncertainty factors which are dependent on the fuel loading pattern and operating conditions, particularly the Traveling In-Core Probe (TIP) reading uncertainty and the R Factor* uncertainty and conclude that the proposed fuel integrity safety limit, a MCPR of 1.06, is acceptable to prevent fuel damage for Quad Cities Units 1 and 2 for fuel cycle 2, the current fuel cycle.

During various transient events, the MCPR will be reduced. To assure that the fuel integrity safety limit (MCPR 1.06) is not exceeded during anticipated transients, the most limiting transients have been analyzed to determine which one results in the largest reduction in critical power ratio (ΔMCPR). This ΔMCPR is added to the safety limit MCPR to establish an operating limit MCPR. CE has submitted the results of analyses of those anticipated transients with the greatest change in MCPR. The most limiting transient analyzed was a turbine trip with failure of the bypass assuming end-of-cycle scram reactivity insertion rates and with reactor power reduced to 90 percent of rated and reactor flow at 100 percent of rated. The change in MCPR for this event is 0.23 for 7 x 7 fuel and 0.29 for 8 x 8 fuel. The resulting limiting condition for operation is a MCPR of 1.29 for 7 x 7 fuel and 1.35 for 8 x 8 fuel.

The proposed technical specifications for Quad Cities 1 included a different limiting condition for operation with respect to MCPR than for Quad Cities 2. CE has stated that the reason for this difference is that the revised dynamic void coefficient used in the analysis of Quad Cities 2 was not used in the Quad Cities 1 analysis. We have concluded that the more conservative revised dynamic void coefficient should be applied to both reactors. Therefore, the MCPR technical specifications proposed for Quad Cities Unit 2 will be issued for Quad Cities Unit 1. This has been discussed with CE.

(3) "General Electric BWR Reload 1 Licensing Submittal for Quad Cities Unit 2," NEDO-20693, December, 1974.

* The R Factor is a parameter which characterizes the local peaking pattern with respect to the most limiting rod.

The assumed end-of-life conditions of 90 percent rated power and 100 percent rated flow are not a conservative set of boundary conditions. A transient from 90 percent of rated power and 90 percent of rated flow would result in a greater Δ MCPR. CE has stated in their letter dated March 27, 1975, that for operation with scram reactivity insertion rates less than that of the generic B curve(4) they will provide a new analysis for the turbine trip with failure of the bypass transient at reduced flow and power conditions. Until such analysis is submitted, this amendment will not authorize operation for Quad Cities Unit 1 when the reactivity insertion rate of the control rods is less than that of the generic B curve presented in Figure 1 of reference 4. This restriction has been included for Quad Cities Unit 2 in Amendment 9. When the analysis for end-of-cycle conditions has been submitted, reviewed, and accepted, the license will be amended accordingly.

In the determination of the operating limit MCPR, the axial power peak is assumed to be at the upper portion of the core (Axial peaking factor of 1.57). We conclude that this assumption is conservative based on the GE study(1) that has shown the operating limit MCPR to be a function of the location of the axial peak and that the largest required MCPR occurs at a location approximately nine feet above the bottom of the twelve foot long fuel. A bottom peaked axial power distribution reduces the required MCPR.

The required operating limit MCPR increases by approximately one percent as the R-Factor decreases from beginning of cycle to end of cycle. However, the R-Factors used are acceptable because at the end of the fuel cycle, the control rods are fully withdrawn and the axial peaking takes place below the core midplane. Therefore, we conclude that the worst consistent set of axial and local peaking factors were used in the analyses.

Finger springs have been attached to the lower end fittings of the reload fuel. The purpose of these springs is to maintain nearly constant bypass flow by restricting deflection of the channel wall during irradiation. A bounding analysis approach was used in the reactor dynamics calculation assuming a 12 percent bypass flow. The uncertainty in this bypass flow was taken into account in the total core flow uncertainty used in the GETAB analysis.

The rod withdrawal error event is discussed in Reference 3 for Quad Cities Unit 2, reload 1, in terms of the worst case conditions. The reports show that the local power range monitor subsystem (LPRMs) will detect high local powers and alarm. However, if the operator

(4) "Dresden Station Special Report 29, Supplement B," dated March 29, 1974.

ignores the LPRM alarm, the rod block monitor (RBM) subsystem will stop rod withdrawal while the critical power ratio is still greater than the 1.06 MCPR safety limit. Therefore, no fuel damage will occur. We conclude that the consequence of this localized event is acceptable.

NEDO-20693 considers loading errors in which an 8 x 8 reload fuel bundle is placed in an improper position. The report states that a loading error accident results in a peak linear heat generation rate (LHGR) of 17.7 kw/ft and a minimum critical power ratio (MCPR) of 1.18 in the misplaced reload fuel bundle. Therefore, no boiling transition occurs. The report also indicates that four fuel bundles adjacent to a misloaded 8 x 8 reload fuel assembly are insignificantly affected by the loading error. We conclude that the consequence of this event is acceptable.

The licensee submitted the results of the recirculation pump seizure accident. The resulting MCPR for this event are 1.18 (for 7 x 7) and 1.29 (for 8 x 8 fuel). This assures that boiling transition does not occur and no fuel cladding damage will occur. We conclude that the consequence of this event is acceptable.

The use of the CPR values in the analysis of the loss-of-coolant accident will be discussed in a separate safety evaluation concerning the reanalysis of the emergency core cooling system which will be issued prior to resumption of reactor operation from the current refueling outage.

Based on the above, we conclude that the operating limit MCPRs proposed by CE are acceptable and that the use and application to Quad Cities Units 1 and 2 of the GEXL transition boiling correlation as a replacement for the Hench-Levy critical heat flux correlation is acceptable. We have modified the proposed GETAB based technical specification with respect to definitions and bases. These modifications have been discussed with Commonwealth Edison and they do not object.

2. <u>APRM Flux Scram and APRM Rod Block Limits</u>

The proposed technical specification changes relating to the APRM flux scram and APRM rod block limits are primarily for the purpose of changing the heat transfer units from heat flux (Btu/hr/ft^2) to rod power (kw/ft). With the conversion to 8 x 8 fuel and to GETAB based technical specifications, this change in units provides a more convenient basis for expressing limits. The proposed changes are associated with maintaining acceptable reactor thermal power and localized fuel power as reactor coolant flow rate (as measured by

recirculation loop flow) changes. The changes affect limits during operations only when local power to average power ratios (total peaking factors) are high. The proposed changes would also clearly specify the limits for the two different types of fuel assemblies in the core (7 x 7 and 8 x 8).

The figures in the technical specifications specifying the limiting safety system settings for APRM flux scram and APRM rod block (Figure 2.1.1) are based on calculations using specified ratios of local to average power. If the actual ratios are higher than the value specified in the reference calculations, the limits are lowered. The way in which these limits are corrected and lowered during operations with high peaking factors is specified by use of Figure 2.1-2 and by the equations in Section 2.1.A.1 and Section 2.1.B of the technical specifications. The corrections are presently stated in terms of peak heat flux and are calculated for 7 x 7 fuel. To clearly state the appropriate correction for 8 x 8 fuel, as well as 7 x 7 fuel, CE has proposed to state the correction in terms of total peaking factors as compared to reference (limiting) total peaking factors. This requires a change in the definition of peaking factor and in the figures and equations which specify the limits.

The proposed limits are specified so that, at 100% power and 100% recirculation flow, the local linear heat generation rate (LHGR) does not exceed the design LHGR 17.5 kw/ft for 7 x 7 fuel, which is not changed, and the corresponding value for 8 x 8 fuel of 13.4 kw/ft.

The specific proposed changes to the technical specifications are itemized below.

Section 1.0 Definitions

Subsection 1.0.K which defines peaking factor in terms of fuel rod surface heat fluxes would be replaced by a new subsection 1.0.BB which defines a total peaking factor in terms of power profile. A new subsection 1.0.J defines a Limiting Total Peaking Factor. These new definitions are needed to be consistent with the revised format of the limits discussed below.

Section 2.1 Fuel Cladding Integrity Limiting Safety System Settings

Subsections A.1 concerning APRM neutron flux scram settings and subsection B concerning APRM rod block settings express the settings in terms of the new definitions of peaking factors rather than in

terms of heat flux and base the required settings on the design linear heat generation rates of 17.5 and 13.4 kw/ft for 7 x 7 and 8 x 8 fuel respectively. Figure 2.1.2 would be changed from a linear plot of peak heat flux versus power to a plot of peak LHGR versus power for 7 x 7 and for 8 x 8 fuel. For power levels between zero and 20% of rated core thermal power, peak LHG rates of 3.5 and 2.68 kw/ft rather than total peaking factors are specified for 7 x 7 and for 8 x 8 fuel respectively. Below these LHGR levels fuel cladding damage because of a departure from nucleate boiling would not be expected and therefore these limiting safety system settings are acceptable.

Section 3.1/4.1 Reactor Protection System Limiting Condition for Operation and Surveillance Requirements

A subsection 3.1.B is proposed which specifies required actions during operation when the Total Peaking Factor is greater than the Limiting Total Peaking Factor. The subsection provides the option of reducing trip settings or adjusting power distribution to conform with Specification 2.1.A.1 or 2.1.B. Subsection 4.1.B revised the surveillance requirement from a daily check of peak heat flux to a daily check of peak LHGR. The changes in 3.1 and 4.1 are consistent with the changes in Section 2.1.

CONCLUSION

Based on our review of item 1 (GETAB) of this evaluation and the considerations discussed therein, we have concluded that there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner. Based on our review of item 2 (APRM Flux Scram and APRM Rod Block Limits) of this evaluation and the considerations discussed therein, we have concluded that because this change 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 and that there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner. We also have concluded, based on the considerations discussed in this evaluation that all of the activities discussed herein will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Date: April 21, 1975

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NOS. 50-254 AND 50-265

COMMONWEALTH EDISON COMPANY AND

IOWA-ILLINOIS GAS AND ELECTRIC COMPANY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

Notice is hereby given that the U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment Nos. 15 and 11 to Facility Operating License Nos. DPR-29 and DPR-30 (respectively) issued to the Commonwealth Edison Company (acting for itself and on behalf of the Iowa-Illinois Gas and Electric Company) which revised Technical Specifications for operation of the Quad Cities Units 1 and 2 located in Rock Island County, Illinois. These amendments are effective as of their date of issuance.

The amendments (1) incorporate operating limits in the Technical Specifications based on the new General Electric Thermal Analysis Basis in accordance with the Commonwealth Edison's request dated December 13, 1974, as supplemented December 20, 1974, January 27, 1975, March 14 and 27, 1975, and April 9, 1975, and (2) authorize changes to the Average Power Range Monitor (APRM) flux scram and APRM Rod Block Limits in accordance with Commonwealth Edison's request dated May 15, 1974, as supplemented October 22 and December 5, 1974.

The applications for these amendments comply 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 amendments. Notice of Proposed Issuance of Amendment to Facility Operating License in connection with item (1) above was published in the FEDERAL REGISTER on February 10, 1975 (40 FR 6240). No request for a hearing or petition for leave to intervene was filed following notice of the proposed action. Prior public notice of item (2) above is not required since the amendment does not involve a significant hazards consideration.

For further details with respect to these actions, see (1) the applications for these amendments dated December 13, 1974 (as supplemented December 20, 1974, January 27, 1975, March 14 and 27, 1975, and April 9, 1975), and May 15, 1974 (as supplemented October 22, 1974 and December 5, 1974), (2) Amendment Nos. 15 and 11 to License Nos. DPR-29 and DPR-30, with Change No. 26, and (3) the Commission's concurrently issued 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, at 504 - 17th Street in Moline,

Illinois 60265. A single 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 Reactor Licensing.

Dated at Bethesda, Maryland, this 21st day of April 1975.

FOR THE NUCLEAR REGULATORY COMMISSION

Dennis L. Ziemann

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