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Docket No. 50-237

Commonwealth Edison Company
ATTN: Mr. R. L. Bolger
Assistant Vice President
Post Office Box 767
Chicago, Illinois 60690

Gentlemen:

The Commission has issued the enclosed Amendment No. 21 to Provisional Operating License No. DPR-19 for Dresden Unit No. 2. The amendment includes changes to the license and its appended Technical Specifications that authorize operation of Dresden Unit No. 2:

(1) with additional 8 x 8 fuel assemblies, in response to your application dated March 15, 1976, and supplements dated April 26 and April 28, 1976, and

(2) using modified operating limits based on an acceptable evaluation model that conforms with Section 50.46 of 10 CFR Part 50, and with limits based on the General Electric Thermal Analysis Basis (GETAB), in response to your applications dated July 1, 1975 and September 3, 1975, and supplements dated July 7, 10, and 21, and August 25, and September 19, 1975, and February 26, March 15, April 6, 9, 19, 26 and 28, and May 17 and 21, 1976.

Copies of the related Safety Evaluation, the Negative Declaration, the Environmental Impact Appraisal and the Federal Register Notice are also enclosed.

Sincerely,

Original signed by
Dennis L. Ziemann
Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors

Enclosures:
See next page

CE was notified on 5/23 that amendment was issued. IaE was notified on 5/24/76 RS

OFFICE ➤	OR:ORB #2	OR:ORB #2	OELD	OR:ORB #2	OR:AD/OR	
SURNAME ➤	RMDiggs	RDSilver:ro	[illegible]	DLZiemann	KRGoller	
DATE ➤	5/21/76	5/21/76	5/21/76	5/22/76	5/23/76	

Form AEC-318 (Rev. 9-53) AECM 0240 ☆ U. S. GOVERNMENT PRINTING OFFICE: 1974-526-166

Enclosures:
1. Amendment No. 21 to
   License No. DPR-19
2. Negative Declaration with
   Impact Appraisal
3. Safety Evaluation
4. Federal Register Notice

cc w/enclosures:
See next page

OFFICE ➤						
SURNAME ➤						
DATE ➤						

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-237

DRESDEN UNIT NO. 2

AMENDMENT TO PROVISIONAL OPERATING LICENSE

Amendment No. 21
License No. DPR-19

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

   A. The applications for amendment by the Commonwealth Edison Company (the licensee) dated July 1, 1975, September 3, 1975 and March 15, 1976, and supplements dated July 7, 10, and 21, and August 25 and September 19, 1975, February 26, 1976, April 6, 9, 19, 26 and 28, and May 17 and 21, 1976, 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 applications, 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. After weighing the environmental aspects involved, 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.F and 3.G of Facility License No. DPR-19 are hereby amended and added (respectively) to read as follows:

OFFICE➤						
SURNAME➤						
DATE➤						

F. Restrictions

Beyond the point in the fuel cycle at which the reactivity reduction rate during a scram is less than that of Curve B in Figure 1 of "Supplement B to Dresden Station Special Report 29," dated March 29, 1974, operation of the reactor shall not exceed the core thermal power versus flow conditions defined by the "Nominal Expected 90% Flow Control Line" on Figure 2.1-3 of the Commonwealth Edison letter (J. S. Abel to Benard C. Rusche) dated June 24, 1975 (Docket No. 50-265).

Beyond the point in the fuel cycle at which the reactivity reduction rate during a scram is less than that of end-of-cycle curve on Figure 1-1 of the Commonwealth Edison letter (J. S. Abel to D. L. Ziemann) dated February 27, 1975 (Docket No. 50-265), operation of the reactor is not authorized.

G. Equalizer Valve Restriction

The valves in the equalizer piping between the recirculation loops shall be closed at all times during reactor operation.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by:
Karl R. Goller

Karl R. Goller, Assistant Director
for Operating Reactors
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: MAY 23 1976

OFFICE➤						
SURNAME➤						
DATE➤						

ATTACHMENT TO LICENSE AMENDMENT NO. 21

PROVISIONAL OPERATING LICENSE NO. DPR-19

DOCKET NO. 50-237

The following changes relate to the Dresden Unit No. 2 Technical Specifications. The changed areas on the revised pages are shown by marginal lines.

Remove Pages	Insert Pages
ii	ii
1 - 18A	1 - 18B
34	34
42 and 42A	42 and 42A
48 and 49	48 and 49
57A	57A
63	63
78	78
81B	81B
81C	81C, 81C-1, 81C-2
81D and 81E	81D - 81G
82	82
84	84
85A and 85B	85A - 85C
86A - 86C	86A - 86D

TABLE OF CONTENTS (Cont)

1.0 DEFINITIONS

The succeeding frequently used terms are explicitly defined so that a uniform interpretation of the specifications may be achieved.

A. (Deleted)

B. Alteration of the Reactor Core - The act of moving any component in the region above the core support plate, below the upper grid and within the shroud. Normal control rod movement with the control rod drive hydraulic system is not defined as a core alteration.

C. 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)

D. Hot Standby - Hot standby means operation with the reactor critical, system pressure less than 600 psig, and the main steam isolation valves closed.

E. Immediate - Immediate means that the required action will be initiated as soon as practicable considering the safe operation of the unit and the importance of the required action.

F. 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 cycle.

G. 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.

H. 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.

I. 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.

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

K. 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 Generation Rate (LHGR) corresponding to the operating limit at 100% power.

L. Logic System Function 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.

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

N. Mode - The reactor mode is that which is established by the mode-selector-switch.

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

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

Q. Operating Cycle - Interval between the end of one refueling outage and the end of the next subsequent refueling outage.

R. 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.

S. Protective Instrumentation Definitions

1. Instrument Channel - An instrument channel means an arrangement of a sensor and auxiliary equipment required to generate and transmit to a trip system a single trip signal related to the plant parameter monitored by that instrument channel.

2. Trip System — A trip system means an arrangement of instrument channel trip signals and auxiliary equipment required to initiate action to accomplish a protective trip function. A trip system may require one or more instrument channel trip signals related to one or more plant parameters in order to initiate trip system action. Initiation of protective action may require the tripping of a single trip system or the coincident tripping of two trip systems.

3. Protective Action — An action initiated by the protection system when a limit is reached. A protective action can be at a channel or system level.

4. Protective Function — A system protective action which results from the protective action of the channels monitoring a particular plant condition.

T. Rated Neutron Flux — Rated neutron flux is the neutron flux that corresponds to a steady-state power level of 2527 thermal megawatts.

U. Rated Thermal Power — Rated thermal power means a steady-state power level of 2527 thermal megawatts.

V. Reactor Power Operation — Reactor power operation is any operation with the mode switch in the "Startup/Hot Standby" or "Run" position with the reactor critical and above 1% rated thermal power.

1. Startup/Hot Standby Mode — In this mode the reactor protection scram trips, initiated by condenser low vacuum and main steamline isolation valve closure, are bypassed when reactor pressure is less than 600 psig; the low pressure main steamline isolation valve closure trip is bypassed, the reactor protection system is energized with IRM neutron monitoring system trips and control rod withdrawal interlocks in service.

2. Run Mode — In this mode the reactor protection system is energized with APRM protection and RBM interlocks in service.

W. Reactor Vessel Pressure — Unless otherwise indicated, reactor vessel pressures listed in the Technical Specifications are those measured by the reactor vessel steam space detector.

X. Refueling Outage — Refueling outage is the period of time between the shutdown of the unit prior to a refueling and the startup of the plant subsequent to that refueling. For the purpose of designating frequency of testing and surveillance, a refueling outage shall mean a regularly scheduled refueling outage; however, where such outages occur within 8 months of the completion of the previous refueling outage, the required surveillance testing need not be performed until the next regularly scheduled outage.

Y. Safety Limit — The safety limits are limits below which the reasonable maintenance of the cladding and primary system are assured. Exceeding such a limit is cause for unit shutdown and review by the Atomic Energy Commission before resumption of unit operation. Operation beyond such a limit may not in itself result in serious consequences but it indicates an operational deficiency subject to regulatory review.

Z. 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.

2. The standby gas treatment system is operable.

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

AA. Shutdown - The reactor is in a shutdown condition when the reactor mode switch is in the shutdown mode position and no core alternations are being performed. When the mode switch is placed in the shutdown position a reactor scram is initiated, power to the control rod drives is removed, and the reactor protection system trip systems are de-energized.

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.

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

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

a. A maximum allowable extension not to exceed 25% of the surveillance interval.

b. A total maximum combined interval time for any 3 consecutive intervals not to exceed 3.25 times the specified surveillance interval.

DD. 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.

EE. 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.

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.

Specifications

A. Reactor Pressure >800 psig 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.

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. APRM Flux Scram Trip Setting (Run Mode) When the reactor mode switch is in the run position, the APRM flux scram setting shall be: $S \leq [.65W + 55]\left[\frac{LTPF}{TPF}\right]$ 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 = 3.05(7X7 fuel assemblies) 3.01(8X8 fuel assemblies) 2. APRM Flux Scram Trip Setting (Refuel 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.

1.1 SAFETY LIMIT	2.1 LIMITING SAFETY SYSTEM SETTING

1.1 SAFETY LIMIT

B. Core Thermal Power Limit (Reactor Pressure $\leq$ 800 psig)

When the reactor pressure is $\leq$ 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.

B. APRM Rod Block Setting

The APRM rod block setting 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. Generator Load Rejection 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 Pressure initiation of main steamline isolation valve closure shall be $\geq$850 psig.
	I. Turbine Control Valve Fast Closure Scram on loss of control oil pressure shall be set at greater than or equal to 900 psig.

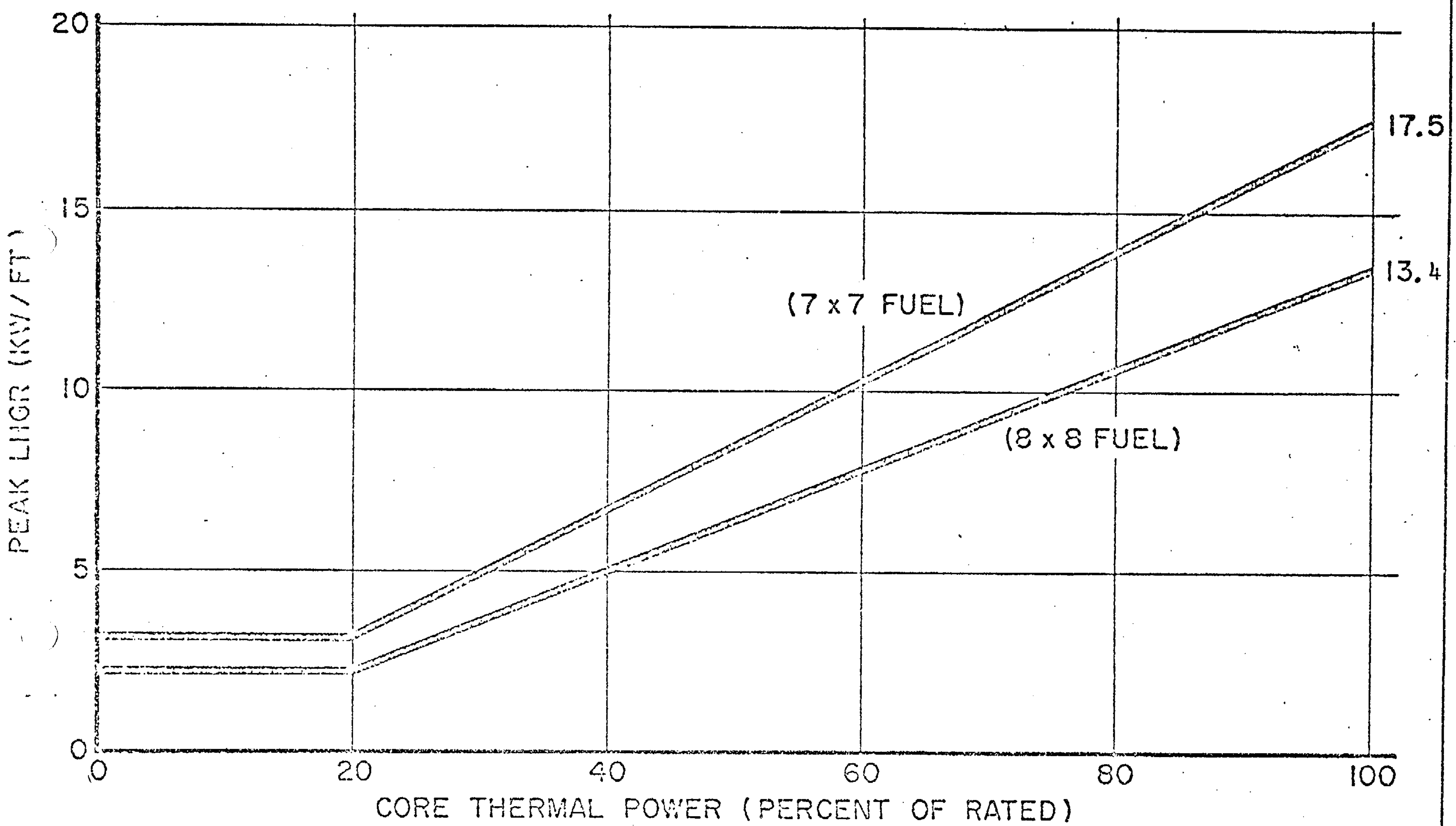

Figure 2.1-2
PEAK LHGR VERSUS CORE THERMAL POWER
FOR A LIMITING TOTAL PEAKING FACTOR

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 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 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

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 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 max which cannot be exceeded under steady power operation.

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(1) NEDO-20694. "General Electric Boiling Water Reactor Reload No. 3 Licensing Submittal for Dresden Nuclear Power Station Unit 3."

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 $28x10^3$ 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 $28x10^3$ lbs/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 3.84 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 each refueling outage and at least every 32 weeks 50% are checked to assume 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 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

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 MWt. In addition, 2527 MWt is the licensed maximum steady-state power level of the units. This maximum steady-state power level will never knowingly be exceeded.

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

2.1 Limiting Safety System Setting Bases

Fuel Cladding Integrity (cont'd)

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. 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 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 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.

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.

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.

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.

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.

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 limited such that MCPR remains above 1.06 even during the worst case transient that assumes the turbine bypass is closed.

F. Generator Load Rejection Scram - The generator load rejection 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 1.06 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 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 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 which occurs when the main steam line isolation valves are closed, to provide for reactor shutdown so that high power operation at low reactor pressure 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 high neutron flux scram. Thus, the combination of main steam line low pressure isolation and isolation valve closure scram 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 anticipates the pressure and flux transients which occur during normal or inadvertent isolation valve closure. With the scrams set at 10% valve closure, there is no increase in neutron flux.

I. Turbine Control Valve Fast Closure Scram

The turbine hydraulic 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 generator load rejection 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 scram on fast closure of the turbine control valves, a scram has been added to the reactor protection system which senses failure of control oil pressure to the turbine control system. This is an anticipatory scram and results in reactor shutdown before any significant increase in neutron flux occurs. The transient response is very similar to that resulting from the generator load rejection. 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 to close until the fluid pressure is 600 psig. Therefore, the scram occurs well before valve closure begins.

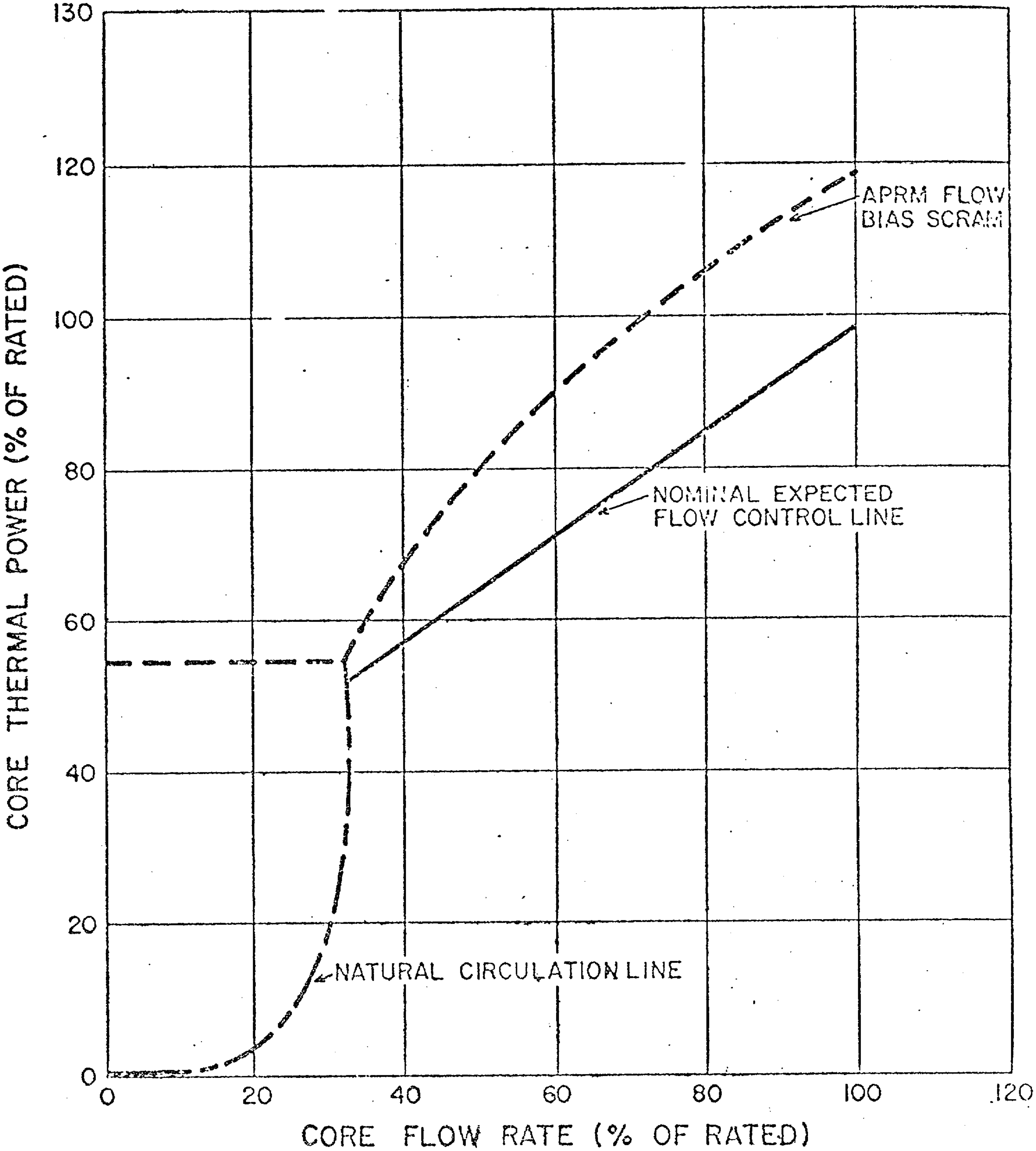

FIGURE 2.1-3 APRM FLOW BIAS SCRAM RELATIONSHIP TO NORMAL OPERATING CONDITIONS

a half scram and rod block condition. Thus, if the calibration were performed during operation, flux shaping would not be possible. Based on experience at other generating stations, drift of instruments, such as those in the Flow Biasing Nework, is not significant and therefore, to avoid spurious scrams, a calibration frequency of each refueling outage is established.

Group (C) devices are active only during a given portion of the operational cycle. For example, the IRM is active during startup and inactive during full-power operation. Thus, the only test that is meaningful is the one performed just prior to shutdown or startup; i.e., the tests that are performed just prior to use of the instrument.

Calibration frequency of the instrument channel is divided into two groups. These are as follows:

1. Passive type indicating devices that can be compared with like units on a continuous basis.

2. Vacuum tube or semiconductor devices and detectors that drift or lose sensitivity.

Experience with passive type instruments in Commonwealth Edison generating stations and substations indicates that the specified calibrations are adequate. For those devices which employ amplifiers, etc., drift specifications call for drift to be less than 0.4%/month; i.e., in the period of a month a drift of .4% would occur and thus providing for adequate margin.

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

A comparison of Tables 4.1.1 and 4.1.2 indicates that six instrument channels have not been included in the latter Table. These are: Mode Switch in Shutdown, Manual Scram, High Water Level in Scram Discharge Tank, Main Steam Line Isolation Valve Closure, Generator Load Rejection, 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. Further, these switches are mounted solidly to the device and have a very low probability of moving, e.g. the switches in the scram discharge volume tank. Based on the above, no calibration is required for these six instrument channels.

B. The LTPF shall be checked once per day to determine if the APRM scram requires adjustment. This may 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.

INSTRUMENTATION THAT INITIATES ROD BLOCK

Minimum No. of Operable Inst. Channels Per Trip System(1)	Instrument	Trip Level Setting
1	APRM upscale (flow bias) (7)	$\leq \left[.65W + 43\right] \left[\frac{LTPF}{TPF}\right]$ (2)
* 1	APRM upscale (refuel and Startup/Hot Standby mode)	$\leq$ 12/125 full scale
2	APRM downscale (7)	$\geq$ 3/125 full scale
1	Rod block monitor upscale (flow bias) (7)	$\leq \left[.65W + 42\right] \left[\frac{LTPF}{TPF}\right]$ (2)
1	Rod block monitor downscale (7)	$\geq$ 5/125 full scale
3	IRM downscale (3)	$\geq$ 5/125 full scale
3	IRM upscale	$\leq$ 108/125 full scale
* 3	IRM detector not fully inserted in the core	
2(5)	SRM detector not in startup position	(4)
2(5)(6)	SRM upscale	$\leq 10^5$ counts/sec

TABLE 3.2.3 (cont)

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, IRM downscale and IRM detector not fully inserted in the core need not be operable in the "Run" position and APRM downscale, APRM upscale (flow bias), 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 seven days provided that during that time the operable system is functionally tested immediately and daily thereafter; if this condition lasts longer than seven days the system shall be tripped. If the first column cannot be met for both trip systems, the systems shall be tripped.

2. W - percent of drive flow required to produce a rated core flow of 98 M lb/hr.

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

4. This function may be 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 when the IRM upscale rod block is operable.

7. Not required while performing low power physics tests at atmospheric pressure during or after refueling at power levels not to exceed 5 MW(t).

Two sensors on the isolation condenser supply and return lines are provided to detect the failure of isolation condenser line and actuate isolation action. The sensors on the supply and return sides are arranged in a 1 out of 2 logic and, to meet the single failure criteria, all sensors and instrumentation are required to be operable. The trip settings of 20 psig and 32" of water and valve closure time are such as to prevent uncovering the core or exceeding site limits. The sensors will actuate due to high flow in either direction.

The HPCI high flow and temperature instrumentation are provided to detect a break in the HPCI piping. Tripping of this instrumentation results in actuation of HPCI isolation valves; i.e., Group 4 valves. Tripping logic for this function is the same as that for the isolation condenser and thus all sensors are required to be operable 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 dual bus system. As for other vital insturmentation arranged in this fashion the Specification preserves the effectiveness of the system 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 1.06. 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.

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.

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.

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.

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.

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.

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

* The rod block which occurs when the IRM detectors are not fully inserted in the core for the refuel and startup/hot standby position of the mode switch has been provided to assure that these detectors are in the core during reactor startup. This, therefore, assures that these instruments are in proper position to provide protection during reactor startup. The IRM's primarily provide protection against local reactivity effects in the source and intermediate neutron range.

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 back-up 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. Section 6.2.6.3 SAR. 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 fifteen 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 stack.

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 instantaneous stack release rate limit given in Specification 3.8 is not exceeded.

Four radiation monitors are provided 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 and on the refueling floor. The trip logic is a 1 out of 2 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 11 mr/hr for the monitors in the ventilation duct are based upon initiating normal ventilation isolation and standby gas treatment system operation to limit the dose

3.3 LIMITING CONDITIONS FOR OPERATION

4. Control rod 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.

5. During operating 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.

4.3 SURVEILLANCE REQUIREMENTS

4. Prior to control rod withdrawal for startup or during refueling verify that at least two source range channels have been 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.

operator with a visual indication of neutron level. This is needed for knowledgeable and efficient reactor startup at low neutron level. The consequences of reactivity accidents are functions of the initial neutron flux. The requirement of at least 3 counts per second assures that any transient, should it occur, begins at or above the initial value of 10^{-8} of rated power used in the analyses of transients 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 are 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 level 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 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. Amendments 17/18 and 19/20 present the results of an evaluation of a rod block monitor failure. These amendments show that 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.

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.

3.5 LIMITING CONDITION FOR OPERATION

D. Automatic Pressure Relief Subsystems

1. Except as specified in 3.5.D.2 and 3 below, the Automatic Pressure Relief Subsystem shall be operable whenever the reactor pressure is greater than 90 psig and irradiated fuel is in the reactor vessel.

2. From and after the date that one of the five relief valves of the automatic pressure relief subsystem is made or found to be inoperable when the reactor is pressurized above 90 psig with irradiated fuel in the reactor vessel, reactor operation is permissible only during the succeeding seven days unless repairs are made and provided that during such time the HPCI Subsystem is operable.

3. From and after the date that more than one of five relief valves of the automatic pressure relief subsystem are made or found to be inoperable when the reactor is pressurized above 90 psig with irradiated fuel in the reactor vessel, reactor operation is permissible only during the succeeding 24 hours unless repairs are made and provided that during such time the HPCI Subsystem is operable.

4.5 SURVEILLANCE REQUIREMENT

D. Surveillance of the Automatic Pressure Relief Subsystem shall be performed as follows:

1. During each operating cycle the following shall be performed:

   a. A simulated automatic initiation which opens all pilot valves, and

   b. With the reactor at low pressure each relief valve shall be manually opened until thermocouples downstream of the valve indicate fluid is flowing from the valve.

   c. A logic system functional test shall be performed each refueling outage.

2. When it is determined that one relief valve of the automatic pressure relief subsystem is inoperable, the HPCI shall be demonstrated to be operable immediately and weekly thereafter.

3. When it is determined that more than one relief valve of the automatic pressure relief subsystem is inoperable, the HPCI subsystem shall be demonstrated to be operable immediately.

I. Average Planar LHGR

During steady state power operation, the average linear heat generation rate (LHGR) of all the rods in any fuel assembly, as a function of average planar exposure, at any axial location, shall not exceed the maximum average planar LHGR shown in Figures 3.5.1-A or 3.5.1-B dependent on fuel type. If at any time during operation it is determined by normal surveillance that the limiting value for APLHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the APLHGR is not returned to within the prescribed limits within two (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.

I. Average Planar Linear Heat Generation Rate (APLHGR)

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

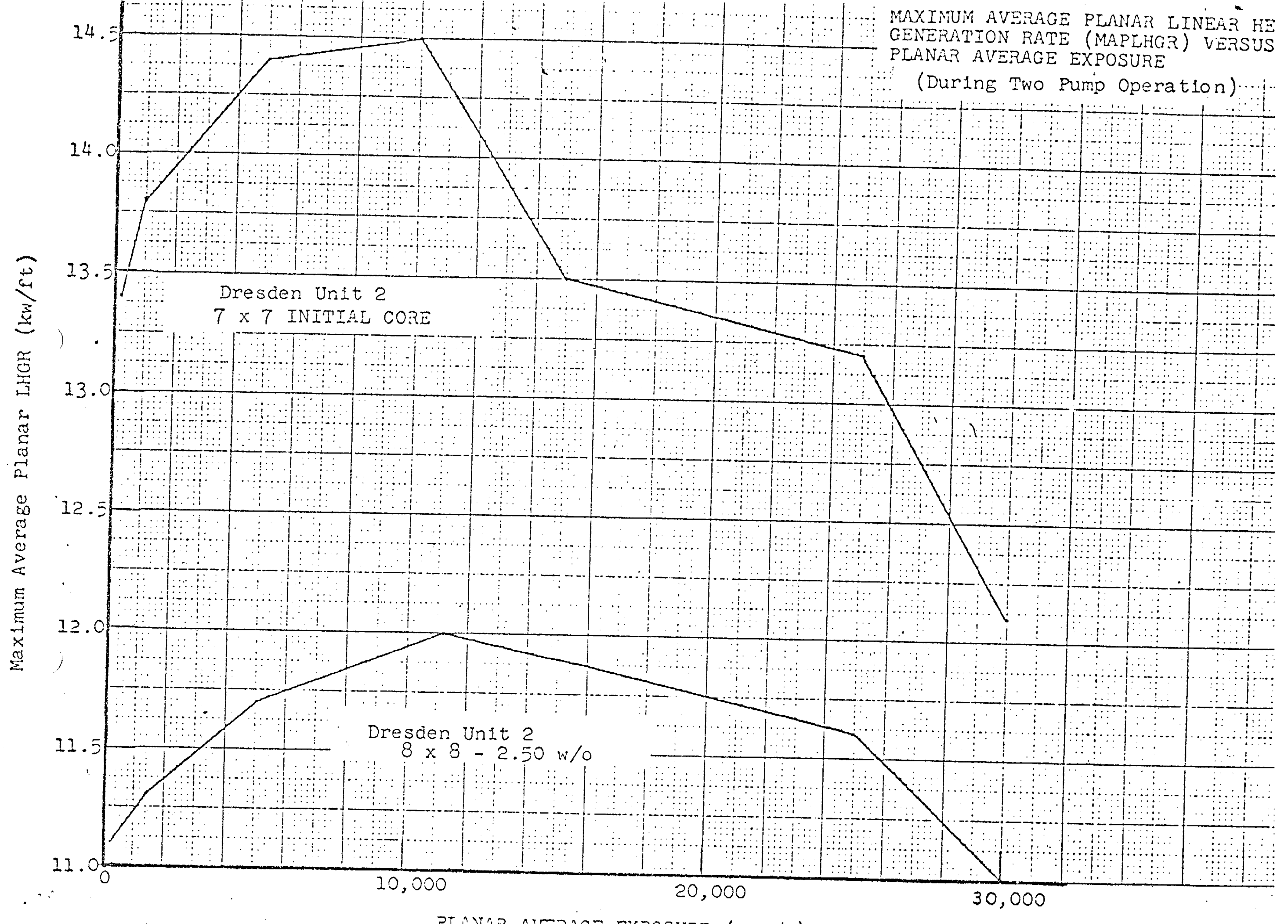
MAXIMUM AVERAGE PLANAR LINEAR HE
GENERATION RATE (MAPLHGR) VERSUS
PLANAR AVERAGE EXPOSURE
(During Two Pump Operation)
Dresden Unit 2
7 x 7 INITIAL CORE
Dresden Unit 2
8 x 8 - 2.50 w/o
Maximum Average Planar LHGR (kw/ft)
14.5
14.0
13.5
13.0
12.5
12.0
11.5
11.0
0
10,000
20,000
30,000
PLANAR AVERAGE EXPOSURE (MWD/T)

FIGURE 3.5.1.A

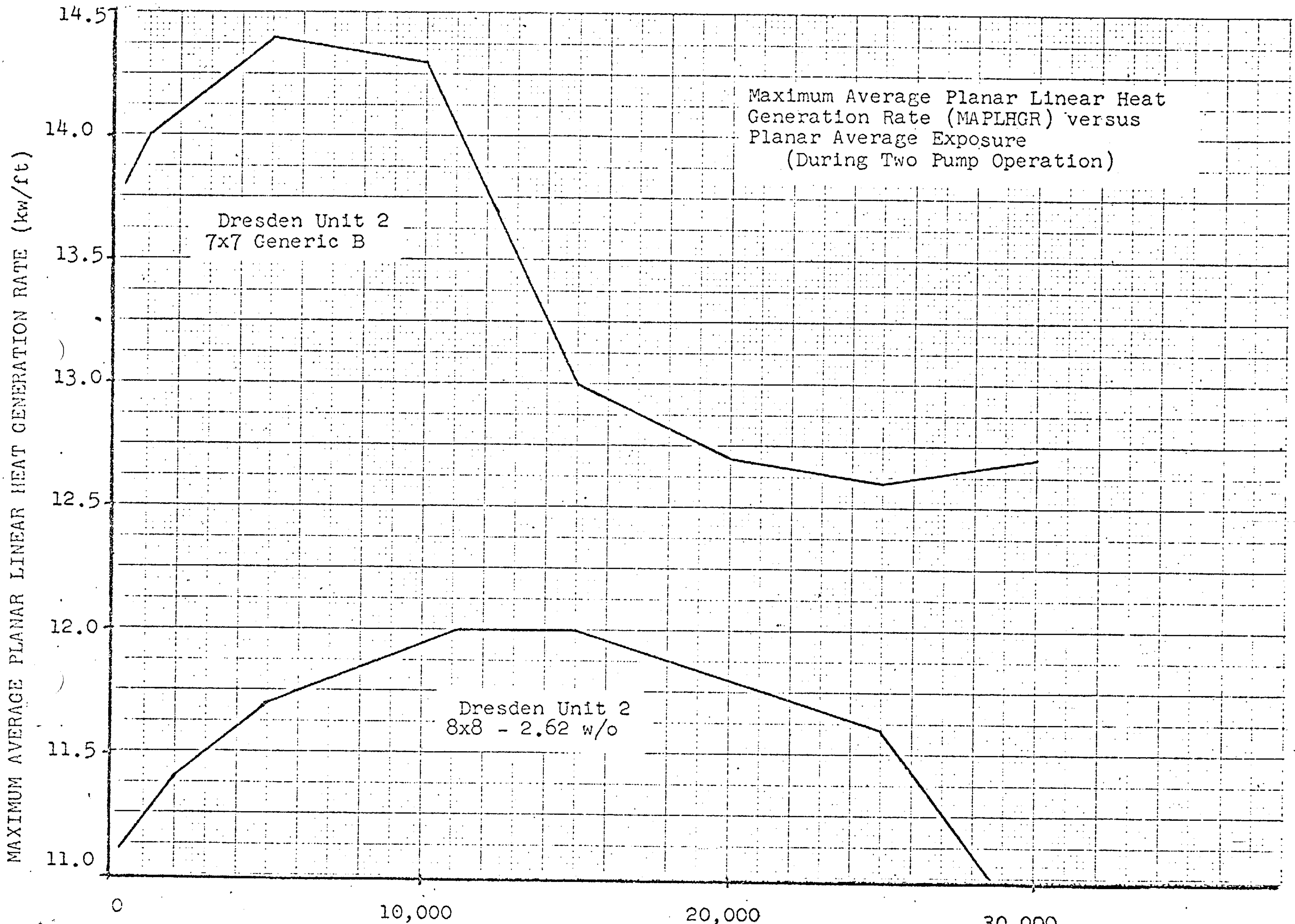
14.5
14.0
13.5
13.0
12.5
12.0
11.5
11.0
MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (kw/ft)
Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) versus Planar Average Exposure (During Two Pump Operation)
Dresden Unit 2
7x7 Generic B
Dresden Unit 2
8x8 - 2.62 w/o
0
10,000
20,000
30,000
PLANAR AVERAGE EXPOSURE (MWD/T)

FIGURE 3.5.1.B

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

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
- .037 initial core fuel
- .026 reload 1, 7x7 fuel
- .022 8x8 fuel

L_T - Total Core Length - 12 ft.

L - Axial distance from bottom of core

If at any time during operation it is determined by normal surveillance that the limiting value for LHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the LHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

J. Local LHGR

The LHGR as a function of core height shall be checked daily during reactor operation at $\geq$ 25% rated thermal power.

3.5 LIMITING CONDITION FOR OPERATION	4.5 SURVEILLANCE REQUIREMENTS

K. Minimum Critical Power Ratio (MCPR)

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

Unit 2

1.29 (7 x 7 fuel)

1.35 (8 x 8 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.

If at any time during steady state power operation, it is determined 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 two (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, the MCPR shall be 1.32 times K_f where K_f is as shown in Figure 3.5-2.

K. Minimum Critical Power Ratio (MCPR)

MCPR shall be determined daily during a reactor power operation at $\geq$ 25% rated thermal power and following any change in power level or distribution that would cause operation with a limiting control rod pattern as described in the bases for Specification 3.3.B.5.

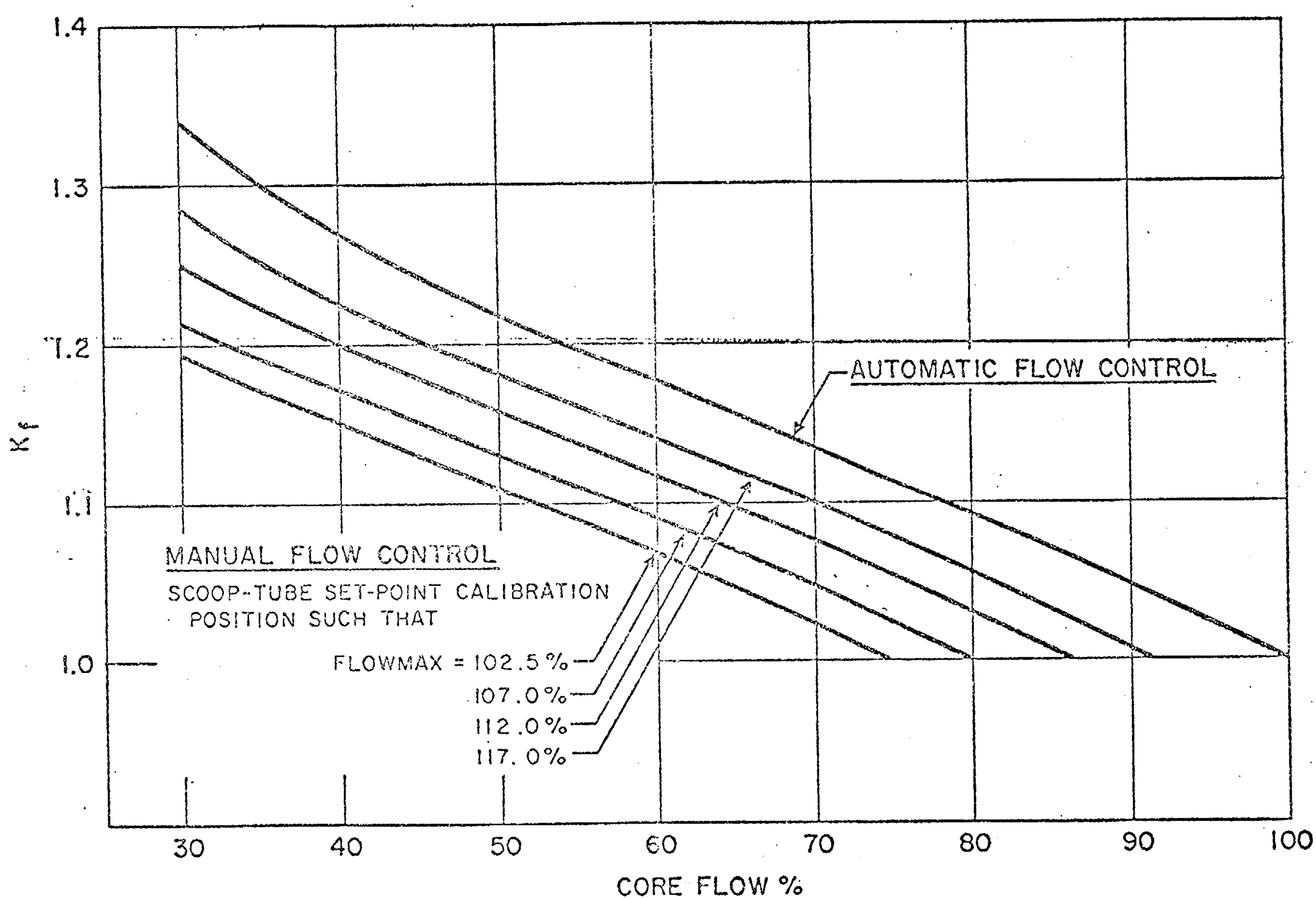

FIGURE 3.5-2 K_f FACTOR

L. Condensate Pump Room Flood Protection

1. The systems installed to prevent or mitigate the consequences of flooding of the condensate pump room shall be operable prior to startup of the reactor.

2. The condenser pit water level switches shall trip the condenser circulating water pumps and alarm in the control room if water level in the condenser pit exceeds a level of 5 feet above the pit floor. If a failure occurs in one of these trip and alarm circuits, the failed circuit shall be immediately placed in a trip condition and reactor operation shall be permissible for the following seven days unless the circuit is sooner made operable.

L. Condensate Pump Room Flood Protection

1. The following surveillance requirements shall be observed to assure that the condensate pump room flood protection is operable.

   a. The testable penetrations through the walls of CCSW pump vaults shall be checked during each operating cycle by pressurizing to 15 ±2 psig and checking for leaks using a soap bubble solution. The criteria for acceptance should be no visible leakage through the soap bubble solution. The bulkhead door shall be checked during each operating cycle by hydrostatically testing the door at 15 ±2 psig and checking to verify that leakage around the door is less than one gallon per hour.

3.5 LIMITING CONDITION FOR OPERATION	4.5 SURVEILLANCE REQUIREMENT
3. If Specification 3.5.K.1 and 2 cannot be met, reactor startup shall not commence or if operating, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours	b. The CCSW Vault Floor drain shall be checked during each operating cycle by assuring that water can be run through the drain line and actuating the air operated valves by operation of the following sensor: i. loss of air ii. high level in the condensate pump room (5'0") c. The condenser pit 5 foot trip circuits for each channel shall be checked once a month. A logic system functional test shall be performed during each refueling outage.

3.5 Limiting Conditions for Operation Bases

A. Core Spray and LPCI Mode of the RHR System - This specification assures that adequate emergency cooling capability is available.

Based on the loss of coolant analyses included in References (1) and (2) in accordance with 10CFR50.46 and Appendix K, core cooling systems provide sufficient cooling to the core to dissipate the energy associated with the loss of coolant accident, to limit the calculated peak clad temperature to 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%.

The allowable repair times are established so that the average risk rate for repair would be no greater than the basic risk rate. The method and concept are described in Reference (3). Using the results developed in this reference, the repair period is found to be less than 1/2 the test interval. This assumes that the core spray and LPCI subsystems constitute a 1 out of 3 system, however, the combined effect of the two systems to limit excessive clad temperatures 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 45 days and this specification is within this period. For multiple failures, a shorter interval is specified and 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 3 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 and the entire LPCI system are available should the

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(1) Dresden Station Special Report No. 40, Supplement A, "Unit 2 and 3 Loss of Coolant Accident Analyses in Conformation with 10CFR50, Appendix K."

(2) NEDO-20566, General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10CFR50 Appendix K.

(3) APED-"Guidelines for Determining Safe Test Intervals and Repair Times for Engineered Safeguards" - April 1969, I.M. Jacobs and P.W. Marriott.

D. Automatic Pressure Relief - The relief valves of the automatic pressure relief subsystem are a back-up to the HPCI subsystem. They enable the core spray or LPCI 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 sprays or LPCI. The core spray and/or LPCI provide sufficient flow of coolant to adequately cool the core.

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 1 relief valve significantly reduces the pressure relief capability and thus a 24-hour repair period is specified.

E. Isolation Cooling System - The turbine main condenser is normally available. The isolation condenser is provided for core decay heat removal following reactor isolation and scram. The isolation condenser has a heat removal capacity sufficent to handle the decay heat production at 300 seconds following a scram. Water will be lost from the reactor vessel through the relief valves in the 300 seconds following isolation and scram. This represents a minor loss relative to the vessel inventory.

The system may be manually initiated at any time. The system is automatically initiated on high reactor pressure in excess of 1060 psig sustained for 15 seconds. The time delay is provided to prevent unnecessary actuation of the system during anticipated turbine trips. Automatic initiation is provided to minimize the coolant loss following isolation from the main condenser. To be considered operable the shell side of the isolation condenser must contain at least 11,300 gallons of water. Make-up water to the shell side of the isolation condenser is provided by the condensate transfer pumps from the condensate storage tank. The condensate transfer pumps are operable from on-site power. The fire protection system is also available as make-up water. An alternate method of cooling the core upon isolation from the main condenser is by using the relief valves and HPCI subsystem in a feed and bleed manner. Therefore, the high pressure relief function and the HPCI must be available together to cope with an anticipated transient so the LCO for HPCI and relief valves is set upon this function rather than their function as depressurization means for a small pipe break.

F. Emergency Cooling Availability - The purpose of Specification D is to assure a minimum of core cooling equipment is available at all times. If, for example, one core spray were out of service and the diesel which powered the opposite core spray were out of service, only 2 LPCI pumps would be available. Likewise, if 2 LPCI pumps were out of service and 2 containment service water pumps on the opposite side were also out of service no containment cooling would be available. It is during refueling outages that major maintenance is performed and during such time that all low pressure core cooling systems may be out of service. This specification provides that should this occur, no work will be performed on the primary system which could lead to draining the vessel. This work would include work on certain control rod drive components and recirculation system. Thus, the specification precludes the events which could require core cooling. Specification 3.9 must also be consulted to determine other requirements for the diesel generators.

3.5 Limiting Condition for Operation Bases (Cont'd)

I. Average Planar LHGR

This specification assures that the peak cladding temperature following a postulated design basis loss-of-coolant accident will not exceed the 2200°F limit specified in 10CFR50 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 LHGR of all the rods in a fuel assembly at any axial location and is only dependent secondarily on the rod to rod power distribution within a fuel 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 10CFR50, Appendix K limit.

The maximum average planar LHGRs shown in Figure 3.5.1 are based on calculations employing the models described in Reference (i). Power operation with LHGRs at or below those shown in Fig. 3.5.1 assures that the peak cladding temperature following a postulated loss-of-coolant accident will not exceed the 2200°F limit. These values represent limits for operation to ensure conformance with 10CFR50 and Appendix K only if they are more limiting than other design parameters. The maximum average planar LHGRs plotted in Fig. 3.5.1 at higher exposures result in a calculated peak clad temperature of less than 2200°F. However the maximum average planar LHGRs are shown on Fig. 3.5.1 as limits because conformance calculations have not been performed to justify operation at LHGRs in excess of those shown.

J. Local LHGR

This specification assures that the maximum linear heat generation rate in any rod is less than the design linear

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(i) Dresden Station Special Report No. 40, Supplement A, "Unit 2 and 3, Loss of Coolant Accident Analyses in Conformance with 10 CFR 50 Appendix K."

heat generation rate even if fuel pellet densification is postulated. The power spike penalty specified is based on that presented in Ref. (2) and assumes a linearly increasing variation in axial gaps between core bottom and top, and assumes with 95% confidence, that no more than one fuel rod exceeds the design LHGR due to power spiking. An irradiation growth factor of 0.25% was used as the basis for determining $\Delta P/P$ in accordance with Refs. (3) and (4).

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 disturbance 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.

For core flow rates less than rated, the steady state MCPR is increased by the formula given in the Specification. This assure 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.

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(2) Fuel Densification Effects on General on General Electric Boiling Water Reactor Fuel," Section 3.2.1, Supplement 6, Aug. 1973.

(3) USAEC Report, "Supplement 1 to the Technical Report on Densification of General Electric Reactor Fuels," Dec. 14, 1973.

(4) GE Planning and Development Memorandum #45, "Length Growth of BWR Fuel Elements", R. A. Proebsthe, October 1, 1973 (Proprietary).

3.5.L Flood Protection

Condensate pump room flood protection will assure the availability of the containment cooling service water system (CCSW) during a postulated incident of flooding in the turbine building. The redundant level switches in the condenser pit will preclude any postulated flooding of the turbine building to an elevation above river water level. The level switches provide alarm and circulating water pump trip in the event a water level is detected in the condenser pit.

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 require- ment for claculating average planar LHGR above 25 per cent rated thermal power is sufficient since power distribution shifts are slow when there have not been signifi- cant 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 con- servative mode relative to MCPR.

The daily requirement for calculating MCPR above 25 percent 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 increase from low flows.

4.5.L Flood Protection

The watertight bulkhead door and the penetration seals for pipes and cables penetrating the vault walls have been designed to withstand the maximum flood conditions. To assure that their installation is adequate for maximum flood conditions, a method of testing each seal has been devised.

To test a pipe seal, another test seal is installed in the opposite side of the penetration creating a space between the two seals that can be pressurized. Compressed air is then supplied to a fitting on the test seal and the space inside the sleeve is pressurized to approximately 15 psi. The outer face of the permanent seal is then tested for leaks using a soap bubble solution.

On completion of the test, the test seal is removed for use on other pipes and penetrations of the same size.

In order to test the watertight bulkhead doors, a test frame must be installed around each door. At the time of the test, a reinforced steel box with rubber gasketing is clamped to the wall around the door. The fixture is then pressurized to approximately 15 psig to test for leaktightness.

Floor drainage of each vault is accomplished through a carbon steel pipe which penetrates the vault. When open, this pipe will drain the vault floor to a floor drain sump in the condensate pump room.

Equipment drainage from the vault doolers and the CCSW pump bedplates will also be routed to the vault floor drains. The old equipment drain pipes will be permanently capped preclude the possibility of back-flooding the vault.

(Cont'd)

4.5 Surveillance Requirement Bases

As a means of preventing backflow from outside the vaults in the event of a flood, a check valve and an air operated valve are installed in the 2" vault floor drain line 6'0" above the floor of the condensate pump room.

The check valve is a 2" swing check designed for 125 psig service. The air operated valve is a control valve designed for a 50 psi differential pressure. The control valve will be in the normally open position in the energized condition and will close upon any one of the following:

Loss of air or power

High level (5'0") in the condensate pump room

Closure of the air operated valve on high water level in the condensate pump room is effected by use of a level switch set at a water level of 5'0". Upon actuation, the switch will close the control valve and alarm in the control room.

The operator will also be aware of problems in the vaults/condensate pump room if the high level alarm on the equipment drain sump is not terminated in a reasonable amount of time. If must be pointed out that these alarms provide information to the operator but that operator action upon the above alarms is not a necessity for reactor safety since the other provisions provide adequate protection.

A system of level switches has been installed in the condenser pit to indicate and control flooding of the condenser area. The following switches are installed:

	Level	Function
a.	1'0" (1 switch)	Alarm, Panel Hi-Water Condenser Pit
b.	3'0" (1 switch)	Alarm, Panel High-Circ. Water Condenser Pit
c.	5'0" (2 redundant switch pairs)	Alarm and Circ. Water Pump Trip

Level (a) indicates water in the condenser pit from either the hotwell or the circulating water system. Level (b) is above the hotwell capacity and indicates a probable circulating water failure.

Should the switches at level (a) and (b) fail or the operator fail to trip the circulating water pumps on alarm at level (b), the actuation of either level switch pair at level (c) shall trip the circulating water pumps automatically and alarm in the control room. These redundant level switch pairs at level (c) are designed and installed to IEEE-279,
23 "Criteria for Nuclear Power Plant Protection Systems." As the circulating water pumps are tripped, either manually or automatically, at level (c) of 5'0", the maximum water level reached in the condenser pit due to pumping will be at the 491'0" elevation (10' above condenser pit floor elevation 481'0"; 5' plus an additional 5' attributed to pump coastdown).

In order to prevent overheating of the CCSW pump motors, a vault cooler is supplied for each pump. Each vault cooler is designed to maintain the vault at a maximum 105°F temperature during operation of its respective pump. For example, if CCSW pump 2B-1501 starts, its cooler will also start and compensate for the heat supplied to the vault by the 2B pump motor keeping the vault at less than 105°F.

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

Operation of the fans and coolers is required during pump operability testing and thus additional surveillance is not required.

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

NEGATIVE DECLARATION

REGARDING PROPOSED CHANGES TO THE TECHNICAL SPECIFICATIONS OF LICENSE DPR-19, DRESDEN NUCLEAR POWER STATION UNIT 2, DOCKET NO. 50-237

The Nuclear Regulatory Commission (the Commission) has considered the issuance of changes to the Technical Specifications of Facility Operating License No. DPR-19. These changes would authorize the Commonwealth Edison Company (the licensee) to operate the Dresden Nuclear Power Station Unit 2 (located in Grundy County, Illinois) with changes to the limiting conditions for operation associated with fuel assembly specific power (average planar linear heat generation rate) which would limit maximum fuel clad temperature in case of a loss of coolant accident, in accordance with the Acceptance Criteria for Emergency Core Cooling System (10 CFR 50.46 and Appendix K to 10 CFR Part 50).

The U. S. Nuclear Regulatory Commission, Division of Operating Reactors has prepared an Environmental Impact Appraisal for the proposed changes to the Technical Specifications of License No. DPR-19, Dresden Unit 2, described above. On the basis of this appraisal, the Commission has concluded that an environmental impact statement for this particular action is not warranted because there will be no environmental impact attributable to the proposed action other than that which has already been predicted and described in the Commission's

Final Environmental Statement for Dresden Nuclear Power Station Units 2 and 3 published in November 1973. The Environmental Impact Appraisal is available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C., and at the Morris Public Library, 604 Liberty Street, Morris, Illinois.

Dated at Bethesda, Maryland, this 21st day of May, 1976.

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by
Dennis L. Ziemann

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

ENVIRONMENTAL IMPACT APPRAISAL BY THE DIVISION OF OPERATING REACTORS

SUPPORTING AMENDMENT NO. 21 TO DPR-19

COMMONWEALTH EDISON COMPANY

DRESDEN NUCLEAR POWER STATION UNIT NO. 2

ENVIRONMENTAL IMPACT APPRAISAL

1. Description of Proposed Action

By letter dated July 1, 1975 and supplements dated July 7, 10 and 21, August 25, September 19, 1975, February 26, 1976, March 15, April 6, 9 and 19, and May 17, and 21, 1976, the Commonwealth Edison Company submitted proposed changes to the Technical Specifications to License No. DPR-19 to incorporate limiting conditions for operation associated with fuel assembly specific power (average planar linear heat generation rate) which would limit maximum fuel clad temperature in case of a loss-of-coolant accident, in accordance with the Acceptance Criteria for Emergency Core Cooling System (10 CFR 50.46 and Appendix K to 10 CFR Part 50). The licensee is at present licensed to possess and operate Dresden Nuclear Power Station Unit No. 2 at power levels up to 2527 MWt. The proposed actions, designed to limit peak fuel clad temperatures in case of a loss-of-coolant accident, are expected to cause no changes in average power level, capacity factor, average fuel failure rate, or total fuel burnup.

2. Environmental Impacts of Proposed Action

The staff has considered the environmental impacts of the proposed action. Since no changes are expected in average power levels or in fuel failure rate under normal operating conditions, there should be no increase in cooling water requirements, thermal effluents, nor in radiological effluents, either liquid or gaseous. The proposed action should, therefore, result in no additional environmental impact on man or on biota in these regards. The principal benefit of electric power production, considered in the benefit-cost analysis of the plant, is unaffected by the action since both the average power level and the fuel burnup are expected to remain the same.

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The proposed action is designed to provide a particular benefit under accident conditions, specifically the loss-of-coolant accident. It will increase the likelihood of minimizing the environmental consequences of the loss-of-coolant accident.

The other environmental impacts considered in the Final Environmental Statement for the Dresden Nuclear Power Station Unit Nos. 2 and 3, Docket Nos. 50-237 and 50-249, November 1973, are not expected to be affected by the proposed action.

3. Conclusion and Basis for Negative Declaration

On the basis of the foregoing analysis, it is concluded that the proposed action will not result in adverse environmental effects in addition to those considered in the Final Environmental Statement for the Dresden Station Unit Nos. 2 and 3. The staff further concludes that a more detailed Environmental Impact Statement is not required for the proposed action, and that a Negative Declaration to this effect is appropriate.

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

Date: MAY 21 1976

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UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-237

COMMONWEALTH EDISON COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO PROVISIONAL OPERATING LICENSE

Notice is hereby given that the U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 21 to Provisional Operating License No. DPR-19 to the Commonwealth Edison Company (the licensee), which revised the license and its appended Technical Specifications for operation of the Dresden Nuclear Power Station Unit No. 2 (the facility) located in Grundy County, Illinois. The amendment is effective as of its date of issuance.

The amendment revised the provisions in the license and its Technical Specifications for the facility to authorize operation (1) with additional 8 x 8 uranium 235 fuel assemblies, and (2) using modified operating limits based on an acceptable evaluation model that conforms with Section 50.46 of 10 CFR Part 50, and with operating limits based on the General Electric Thermal Analysis Basis (GETAB), in accordance with the licensee's applications for the amendment as referenced in the last paragraph of this notice.

The applications for the amendment 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

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in 10 CFR Chapter I, which are set forth in the license amendment. Notice of Proposed Issuance of Amendment to Provisional Operating License in connection with item (2) above was published in the FEDERAL REGISTER on December 2, 1975 (40 FR 55908). No request for a hearing or petition for leave to intervene was filed following notice of the proposed action on item (2) above. Prior public notice of item (1) above was not required since the action does not involve a significant hazards consideration. In connection with the action on Section 50.46 regarding emergency core cooling system (part of item 2) the Commission has issued a Negative Declaration and Environmental Impact Appraisal. In connection with the action identified as item (1) of this Notice, the Commission has determined that the action will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental statement, negative declaration or environmental impact appraisal need not be prepared.

For further details with respect to this action, see (1) the applications for amendment dated July 1, 1975, September 3, 1975, March 15, 1976, and supplements dated July 7 and 10, August 25, September 19, 1975, February 26, 1976, April 6, 9, 19, 26 and 28, and May 17 and 21, 1976, (2) the April 8, 1976 Quad Cities Unit No. 2 licensee submittal in Docket No. 50-265 which is applicable to Dresden 2 and is the non-proprietary version of the Dresden 2 proprietary submittal dated July 21, 1975, (3) Amendment No. 21 to License No. DPR-19, (4) the Commission's concurrently issued related

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Safety Evaluation, and (5) the Commission's Negative Declaration dated (which is also being published in the FEDERAL REGISTER) and associated Environmental Impact Appraisal. 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 Morris Public Library at 604 Liberty Street in Morris, Illinois 60451. A single copy of items (2) through (5) 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 23rd day of May, 1976.

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by
Dennis L. Ziemann

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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 21 TO FACILITY OPERATING LICENSE NO. DPR-19

COMMONWEALTH EDISON COMPANY

DRESDEN UNIT NO. 2

DOCKET NO. 50-237

1.0 INTRODUCTION

Commonwealth Edison Company has proposed to operate Dresden Unit 2:

(1) with additional 8 x 8 fuel assemblies, as requested in their application dated March 15, 1976, and supplements dated April 26 and April 28, 1976;

(2) using modified operating limits based on an acceptable emergency core cooling system evaluation model that conforms with Section 50.46 of 10 CFR Part 50, and with operating limits based on the General Electric Thermal Analysis Basis (GETAB), as requested in their application dated July 1, 1975, and supplements dated July 7, 10 and 21, 1975, and August 25, and September 3 and 19, 1975, and Feburary 26, March 15, April 6, 9, 19, 26 and 28, and May 17 and 21, 1976.

2.0 RELOAD

The licensee has proposed to reload Dresden Unit 2 for Cycle Number 5 with as many as 80, 8 x 8 fuel assemblies having an average enrichment of 2.50 wt% U-235 and 80, 8 x 8 fuel assemblies having an average enrichment of 2.62 wt% U-235. Due to previous operations, this cycle is labeled #5 although it is only the second reload. The documentation submitted in support of the proposed reload consisted of the GE BWR Reload-2 Licensing Submittal for Dresden Unit 2, NEDO - 21145 (1) (2) the GE BWR Generic Reload Application for 8 x 8 fuel, NEDO - 20360 (3); and the proposed Technical Specification changes. (4) The staff has reviewed the submitted information and report our safety evaluation herein.

2.1 Nuclear Characteristics

The information presented in the licensing submittal for the reconstituted core closely follows the guidelines of Appendix A of Reference 3. The licensee relies heavily on this licensing topical report, NEDO - 20360 (3). Although later supplements to this report are undergoing review by the staff, this topical has been found acceptable for use for reactors containing 8 x 8 reload fuel. Up to 160 8 x 8 reload fuel bundles will be loaded throughout the core. As many as 80 of these reload fuel bundles will have an average enrichment of 2.50% by weight of the Uranium-235 isotope while the remainder, as many as 80 fuel bundles, will have an average enrichment of 2.62%. The core contains a total of 724 fuel bundles. Thus, about 22 percent of the fuel bundles are being replaced for this reload. Previously for Reload 1, 32 fuel bundles in 7 x 7 configuration of average enrichment of 2.30% and 124 fuel bundles in 8 x 8 configuration of average enrichment 2.50% had been loaded. The loading pattern for Cycle 5 may be described as follows: (1) the two rows and columns of fuel bundles which intersect at the center of the core will not contain any Reload 2 fuel, (2) the lower enrichment reload bundles are loaded in the interior of the core while the higher enrichment reload bundles are loaded near the outer periphery of the core, (3) in the core interior only one fuel bundle in a four bundle array surrounding a control rod will be replaced, (4) near the core periphery two diagonally located fuel bundles of the four bundle array surrounding a control rod will be replaced, and (5) some initial fuel bundles will be shuffled. The 8 x 8 reload fuel for the Cycle 5 core are, therefore, basically scatter loaded. The data in Reference 1 indicate that the nuclear characteristics of the Reload 2, 8 x 8 fuel bundles are similar to those previously loaded. Thus, the total control system worth, temperature, and void dependent behavior of the reconstituted core will not differ significantly for those values which were previously reported for Dresden Unit 2.

The shutdown margin of the reconstituted core meets the Technical Specification requirement that the core be at least 0.25%Δk subcritical in the most reactive operating state with the largest worth control rod fully withdrawn and with all other control rods fully inserted. The minimum shutdown margin occurs at the beginning of Cycle 5. The analysis considered the fresh fuel assemblies loaded into the core at the beginning of Cycle 5. The assemblies from Cycle 4 remaining in the core were calculated to have an average exposure of 11,070 Mwd/t at the end of Cycle 4.

The information presented in Reference 1 indicates that a boron concentration of 600 ppm in the moderator will make the reactor subcritical by a least 0.03Δk at 20°C, xenon free. Therefore, the requirement for an alternate shutdown system is met by the Standby Liquid Control System.

The Technical Specification requirement for the storage of fuel for Dresden Unit 2 is that the effective multiplication factor, k_{eff}, of the fuel as stored in the fuel storage rack is equal to or less than 0.90 for normal conditions. This is achieved if the uncontrolled k_{eff} of a single fuel bundle is less than 1.30[(3)] at 65°C. The 8 x 8 8D250 and 8D262 fuel bundles, at both the zero exposure and the peak reactivity point, have a k_{eff} less than 1.26 and, therefore, meet the fuel storage requirement for Dresden Unit 2.

Current estimates indicate that the actual plant scram reactivity curve is better than the GE generic "B" curve used in the transient analyses of Reference 5 until approximately 550 MWd/t before the end of Cycle 5. For the reference core loading, this is approximately 3500 MWd/t into the cycle. The end-of-cycle curve used in Reference 5 is more conservative than the predicted end-of-cycle curve for Cycle 5 as shown in figure 1 of Reference 2. Thus, for conservative application of the transient analyses of Reference 5, the "B" scram curve is assumed for Cycle 5 for the first 3500 MWd/t of exposure while the end-of-cycle curve of Reference 5 is applicable for the remainder of the cycle. These scram curves are multiplied by a design conservatism factor of 0.8 for use in the anticipated transient analyses.

The void and Doppler coefficients of reactivity for Cycle 5 are given in Table 5-1 of Reference 1. The void coefficient of reactivity at the core average void fraction of 34 percent varies from -10.3 to -11.5 x 10^{-4} $\Delta k/k/\Delta\%V$. The Doppler coeffieient of reactivity at a fuel temperature of 650°C varies from -1.165 to -1.226 x 10^{-5} $\Delta k/k/\Delta T$. Also, the effective delayed neutron fraction varies from 0.00548 to 0.00603 over the fuel cycle.

Thus, based on our review of the information presented in the Dresden Unit 2 licensing submittal, and the generic 8 x 8 reload report (Reference 3), we conclude that the nuclear characteristics and performance of the reconstituted core for the Reload 2 fuel cycle will not differ significantly from previous Dresden Unit 2 fuel cycles.

2.2 Mechanical Design

The Reload 2 fuel has the same mechanical configuration and fuel bundle enrichments as the 8D250 and 8D262 fuel assemblies described in the 8 x 8 generic reload report (Reference 3). This generic report has been reviewed and with some modifications was found acceptable for use for reactors containing 8 x 8 reload fuel, when supplemented with plant specific information required by our status report (Reference 6) on the GE generic report evaluation. Mechanical and operating parameters for the 8 x 8 assemblies are compared to the 7 x 7 assemblies in Table I. The smaller diameter rods, with lower linear heat generation rate and increased cladding thickness/diameter ratio for the 8 x 8 fuel design as compared to the 7 x 7 fuel assemblies, result in increased safety margins with respect to maximum design linear heat generation rate. The 8D250 Reload 2 fuel incorporates finger springs for

TABLE 1

COMPARISON OF PARAMETERS FOR 8 x 8 AND 7 x 7

ROD FUEL ASSEMBLY DESIGN

	7 x 7	8 x 8
Pellet Outside Diameter (in.)	0.477	0.416
Rod Outside Diameter (in.)	0.563	0.493
Rod-to-Rod Pitch (in.)	0.738	0.640
Water-Fuel Ratio (cold)	2.53	2.60
Weight of U in Assembly (pounds)	412.8	404.6
Cladding Thickness (mils)	37	34
Active Fuel Length (in.)	144	144

controlling moderator/coolant bypass flow at the interface of the channel and fuel bundle lower tie plate. This device has been used satisfactorily in General Electric's initial core and reload fuel for all BWR-4 plants, and for several BWR-3 plants. The finger springs employed in Dresden 2 Reload 2 fuel are identical in design to those that have been used previously on Dresden Unit 3, and Quad Cities, Unit 2 and Unit 3, which are similar plants. Inspection of more than 900 fuel assemblies in operating plants employing finger springs has not revealed any problems relating to their use.

On the basis of our review of the generic 8 x 8 reload report and, current operating experience with the 8 x 8 reload design in similar plants, we conclude that the Dresden Unit 2 Reload 2 Mechanical Design is acceptable.

2.3 Thermal-Hydraulics

The GE generic 8 x 8 fuel reload topical report (3) and GETAB (7) are referenced to provide the description of the thermal-hydraulic methods which were used to calculate the thermal margins. Application of GETAB 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 the accident analyses are satisfied.

We have evaluated and report in the following sections the Dresden Unit 2 Cycle 5 thermal margins based on the GETAB report [7] and plant specific input information provided by the licensee.

2.3.1 Fuel Cladding Integrity Safety Limit MCPR

The fuel cladding integrity safety limit MCPR is 1.06. It is based on the GETAB statistical analysis which assures that 99.9% of the fuel rods in the core are not expected to experience boiling transition during anticipated operational transients. The uncertainties in the core and system operating parameters and the GEXL correlation, (Table 4-1 of the licensee submittal [1]), combined with the relative bundle power distribution in the core form the basis for the GETAB statistical determination of the safety limit MCPR. The tabulated list of uncertainties for Dresden Unit 2 Cycle 5 are the same or more conservative than those used in GETAB (revision of Table IV-1 of NEDO-10958 (Reference 8)). For example, the standard deviation for the TIP readings uncertainty for the subject reload is 8.7% whereas the GETAB NEDO - 10958 report shows 6.3%. The increase in uncertainty for the subject reload is a consequence of the increased uncertainty in the measurement of power in a reload core. A TIP uncertainty of 6.3% would be applicable if this were the initial core. In both cases the TIP reading uncertainties are based on a symmetrical planar power distribution.

The generic core selected for the GETAB statistical analysis is a typical 251/764 core while the Dresden Unit 2 is a 251/724 core. The generic GETAB statistical analysis results are conservative since the bundle power distribution used for the GETAB application has more high power bundles than the distribution expected during the fifth cycle of operation of the Dresden Unit 2 reactor. This results in a conservative value of the MCPR which meets the 99.9% criterion. We conclude that the proposed fuel integrity safety limit, a MCPR of 1.06 is acceptable for Dresden Unit 2 Fuel Cycle 5.

2.3.2 Operating Limit MCPR

Various transient events will reduce the MCPR below the operating MCPR. To assure that the fuel cladding integrity safety limit (MCPR of 1.06) is not exceeded during anticipated operational transients, the most limiting transients have been analyzed to determine which one results in the largest reduction in critical power ratio (ΔMCPR). The licensee has submitted the results of analyses of those transients which produce a significant decrease in MCPR (References 1 and 2). Among the transients evaluated were overpressure, feedwater temperature decrease, coolant flow increase and rod withdrawal error. The most limiting transient is the turbine trip without bypass assuming end of cycle (EOC) scram reactivity insertion rates assuming 90% of rated power and 100% of rated flow . The turbine trip transient results in ΔMCPR's of 0.23 (7 x 7 fuel) and 0.29 (8 x 8 fuel). Addition of these ΔMCPR's to the safety limit MCPR (1.06) gives the minimum operating limit MCPR for each fuel type required to avoid violation of the safety limit, should this limiting transient occur. Therefore, the operating limit MCPR's are 1.29 for 7 x 7 fuel and 1.35 for 8 x 8 fuel. The calculated change in MCPR for the second most severe anticipated transient, the loss of feedwater heating, is 0.15 for 7 x 7 fuel and 0.17 for 8 x 8 fuel.

The transient analyses were evaluated with scram reactivity insertion rates that included a design conservatism factor of 0.80. The design conservatism factor for the void coefficient used was 1.33 and the design conservatism factor for Doppler coefficient was 0.90. The initial conditions(1) used for the worst operational transient are acceptable. The initial MCPR assumed in the transient analyses was equal to or greater than the established operating limit MCPR of 1.35.

A GE study (7) has shown that the required operating MCPR varies with the axial and local power peaking distribution. Axial peaking in the middle or upper portion of the core results in higher required MCPR's than peaking in the lower portion of the core. In the analyses the axial power peak was assumed to be representative of beginning-of-cycle conditions and to be located in the upper portion of the core.

The R-factors, which are a function of the local power peaking, assumed in the analyses are also representative of beginning-of-cycle conditions. The values assumed are 1.075 for 7 x 7 fuel and 1.102 for 8 x 8 fuel. During the cycle the local peaking and therefore the R-factor is reduced while the peak in the axial shape moves toward the bottom of the core. Although the operating limit MCPR would be increased by approximately 1% by the reduced end-of-cycle R-factor, this is offset by the reduction in MCPR resulting from the relocation of the axial peak to below the midplane.

It is concluded from the analyses of the limiting pressure transient, a generator load rejection with bypass failure, that Dresden Unit 2 Cycle 5 can operate at 100% power until that point in the fuel cycle (approximately 3500 MWd/t into the cycle) when the scram reactivity is less than that of the "B" curve in Figure 1 of Reference 5. The power will then be limited to 90% of rated power at 100% of rated flow for the remainder of the cycle.[4] The flow control line is shown on the power/flow map appearing in Figure 1 of Reference 9. Since the transient and safety analyses with a reduced scram reactivity insertion rate are based on the power/flow line defined by the 90% power/100%

flow, operation above this line could result in calculated transients that violate the MCPR and pressure safety limits. Therefore, in accordance with the licensee's proposal, Reference 4, operation is restricted to power/flow conditions along or below this derated flow control line which is consistent with the rod patterns necessary to give the derated power levels at 100% flow.

Conservatism was applied in the determination of the required operating limit MCPR because the assumed axial and local peaking were representative of the beginning of the fuel cycle. This is the worst consistent set of axial and local peaking.

Analyses have shown that the operating limit MCPR's of 1.29 for 7 x 7 fuel and 1.35 for 8 x 8 fuel assure that the fuel cladding integrity safety limit is not exceeded during ancitipated abnormal operational transients. Hence we conclude that the operating limit MCPR's of 1.29 for 7 x 7 fuel and 1.35 for 8 x 8 fuel are acceptable.

2.3.3 <u>Rod Withdrawal Error</u>

The rod withdrawal error transient is discussed in Reference 1 in terms of worst case conditions. Assumptions and descriptions of the rod withdrawal event are given in Reference 3. The information in these two references indicates that the local power range monitor subsystem (LPRM's) will detect high local powers and alarm. However, if the operator ignores the LPRM alarm, the rod block monitor subsystem (RBM) will stop the rod withdrawal while the critical power ratio is still equal to or greater than the 1.06 MCPR safety limit and the cladding is under the one percent plastic strain limit. This rod withdrawal error transient is not limiting for the Dresden Unit 2 Cycle 5 with the RBM setting at 110% of its initial level. We conclude that the analysis performed for this localized transient and the consequences of this localized transient are acceptable.

2.3.4 Operating MCPR Limits for Less Than Rated Power and Flow

The most limiting transient during operation at less than rated flow and power may be different than the limiting transients at rated flow and power. In addition, the operating MCPR needed to assure that the fuel cladding integrity safety limit is not exceeded during anticipated operational transients may be different than the operating MCPR needed at rated flow and power. The limiting transient at lower than rated power and flow conditions is recirculation pump speed control failure. To assure that the fuel cladding integrity safety limit is not exceeded during this limiting transient, Technical Specification limiting conditions for operation (Figure 3.5-2 - page 81E of the Technical Specifications) will require that for core flows less than the rated flow, the licensee will maintain the MCPR greater than the operating minimum values (1.29 for 7 x 7 fuel and 1.35 for 8 x 8 fuel). The minimum MCPR values for less than rated flow are the rated flow values multiplied by the respective K_f factors appearing in Figure 3.5-2 of the Technical Specifications. The K_f factor curves were generically derived and assure that the most limiting transient occurring at less than rated flow will not exceed the safety limit MCPR of 1.06. We conclude that the calculated consequences of the anticipated transients do not violate the thermal and plastic strain limits of the fuel or the pressure limits of the reactor coolant boundary.

2.4 Accident Analysis

2.4.1 ECCS Appendix K Analysis

2.4.1.1 ECCS Performance

On December 27, 1974, the Atomic Energy Commission issued an Order for Modification of License implementing the requirements of 10 CFR 50.46 "Acceptance Criteria and Emergency Core Cooling Systems for Light Water Nuclear Power Reactors." One of the requirements of the Order was that prior to any license amendment authorizing any core reloading" ...the licensee shall submit a re-evaluation of ECCS cooling performance calculated in accordance with an acceptable evaluation model which conforms to the provisions of 10 CFR Part 50, 50.46." The order also required that the evaluation shall be accompanied by such proposed changes in Technical Specifications or license amendments as may be necessary to implement the evaluation results.

On March 15, 1976, via Reference 4 the licensee submitted an amendment to the facility operating license, requesting changes to the Technical Specifications for Dresden Unit 2 to implement the results of their ECCS evaluation. Reference 10 states that the ECCS evaluation of the lead plant (Quad Cities Unit 2) as presented in references 11 and 12 is strictly applicable to Dresden Unit 2. These analyses showed compliance to the 10 CFR 50.46 criteria and Appendix K to 10 CFR Part 50. The Order for Modification of License issued December 27, 1974, stated that evaluation of ECCS cooling performance may be based on the vendor's evaluation model as modified in accordance with the changes described in the staff Safety Evaluation Report of Dresden Unit 2, dated December 27, 1974.

The background of the staff review of the General Electric ECCS model and its application to Dresden Unit 2 is described in the staff Safety Evaluation Report (SER) for these facilities dated December 27, 1974 (The December 27, 1974 SER) issued in connection with the Order. The bases for acceptance of the principal portions of the evaluation model are set forth in the staff's Status Report of October 1974 which are referenced in the December 27, 1974 SER. The December 27, 1974 SER also describes the various changes required in the earlier GE evaluation model. Together, the December 27, 1974 SER and the Status Report and its Supplement, describe an acceptable ECCS evaluation model and the basis for the staff's acceptance of the model. The Dresden Unit 2 evaluation which is covered by this SER properly conforms to the accepted model.

With respect to reflood and refill computations, the Dresden Unit 2 analysis was based on a modified version of the SAFE computer code, with explicit consideration of the staff recommended limitations. These are described on pages 7 and 8 of the December 27, 1974 SER. The Dresden Unit 2 evaluation did not attempt to include any further credit for other potential changes which the December 27, 1974 SER indicated were under consideration by GE at that time.

During the course of our review, we concluded that additional individual break sizes should be analyzed to substantiate the break spectrum curves submitted in connection with the evaluation provided in August 1974. We also requested that other break locations be studied to substantiate that the limiting break location wasin the recirculation line.

The additional analyses [(12)] supported the earlier submittal which concluded that the worst break was the complete severence of the recirculation line. These additional calculations provided further details with regard to the limiting location and size of break as well as worst single failure for the Dresden Unit 2 design. The limiting break which is the design basis accident is the complete severence of the recirculation suction line assuming a failure of the LPCI injection valve.

We have reviewed the evaluation of ECCS performance submitted by Commonwealth Edison for Dresden Unit 2 and conclude that the evaluation has been performed wholly in conformance with the requirements of 10 CFR 50.46(a). Therefore, operation of the reactor would meet the requirement of 10 CFR 50.46 provided that operation is limited to the maximum average planar linear heat generation rates (MAPLHGR) of figures 3.5.1, 3.5.1A and 3.5.1B of Commonwealth Edison's March 15, 1976 submittal (Reference 4) and to a minimum critical power ratio (MCPR) greater than 1.18.

However, certain changes must be made to the proposed Technical Specifications to conform with the evaluation of ECCS performance. The largest recirculation break area assumed in the evaluation was 4.2 square feet. This break size is based on operation with a closed valve in the equalizer line between the two recirculation loops. Therefore, reactor operation will be prohibited unless the valve in the equalizer line is closed.

The ECCS performance analysis assumed that reactor operation is limited to a MCPR of 1.18. However, the operating MCPR limit is more limiting. The Technical Specifications will require that operation in excess of the limiting MCPR values must be reported as a reportable occurrance, even if corrective action is taken upon discovery.

An evaluation was not provided for ECCS performance during reactor operation with one recirculation loop out of service. Therefore, continuous reactor operation under such conditions is not authorized until the necessary analyses have been performed, evaluated and determined acceptable. Operation with one recirculation loop out of service is not allowed for more than 24 hours.

The LOCA analysis assumed all ADS valves operated for small line breaks with HPCI failure. Since the licensee did not provide a LOCA analysis with one ADS valve out of service for small line breaks, the Technical Specifications will not allow continuous operation with any ADS valve out of service, except that one valve may be inoperable for up to seven days.

4.1.2 SINGLE FAILURE CRITERION EVALUATION

Appendix K of 10 CFR 50 requires that an analysis be performed of possible failure modes of ECCS equipment and their effects on ECCS performance and that in carrying out the accident evaluation, the combination of ECCS subsystems assumed to be operating shall be those available subsequent to the most damaging single failure of ECCS equipment. We require that the single failure analysis include electrical, instrumentation and control failure. The licensee has identified the most damaging single failure as the failure of the LPCI injection valve to the unbroken recirculation loop to open. We performed a review to assure that there are no single failures of greater consequences.

The licensee responded to requests for information which were designed to identify single failure problems in equipment necessary for emergency core cooling.

A review of this information identified four areas that required special consideration. These areas are discussed below:

1. Single Failure of Valves

The licensee performed a study to determine the effects of a single failure or operator error that could cause a manually controlled, electrically operated valve in the ECCS system to move to an undesirable position. The conclusion of this study is that no single failure of valves can be postulated that produces an effect with greater consequences on the ECCS than the failure to open of the LPCI valve in the unbroken loop.

The staff has reviewed the ECCS and agrees with the conclusion of the licensee that no design changes are necessary to meet the single failure criterion for manually controlled, electrically operated valves.

2. Submerged Electrical Equipment

The licensee documented that no safety related equipment becomes submerged as a result of a LOCA.

Non-safety related equipment which could become submerged following a LOCA are the drywell floor and equipment drain sump pumps. The loss of these pumps is of no safety significance. There is however, a concern due to their attachment to safety related buses. The licensee has verified that these pumps are prevented from operation (power is prevented from being applied) by interlocks with isolation valves which close on a LOCA signal. The safety related buses are protected by this action which occurs prior to pump submergence. Secondarily, breaker protection has been provided to clear any faults should the interlock fail.

It is the staff's conclusion that the submergence of these pumps is not of safety significance and that the assumption of single failure after submergence does not produce effects of greater consequence than the most damaging single failure.

3. Automatic Transfer of Buses

The present design for the LPCI system includes an automatic transfer feature between redundant power sources. The purpose of this design feature is to enable the ECC system to meet the single failure criterion. It is the staff's position that the swing bus concept currently employed for the LPCI valves is not an acceptable long term solution for meeting the single failure criterion. The present design includes a redundant breaker scheme whose coordination with the supply bus breaker gives a low probability of occurrence that a fault on the swing bus will propagate to the supply bus. The automatic transfer scheme is designed with redundant interlocks such that a single failure will not tie the redundant buses together. The staff concludes that this is acceptable on a short term basis. For the long term, the staff requested that the licensee provide a design for the ECCS which meets the single failure criterion without reliance upon automatic transfer schemes between redundant power sources. The licensee has agreed[10] to (1) evaluate the impact on plant safety of the removal of the automatic transfer scheme and (2) propose and implement a design change on the swing bus that meets the requirements of IEEE Std 308-1971 and the recommendations of Regulatory Guide 1.6, or submit additional justification for the present design. The staff will review this item and assure that it has been resolved prior to Cycle 6 operation.

4. Equipment Qualification

The staff had previously reviewed and accepted the environmental qualification of equipment required for ECCS within containment for the accident environment. The licensee has stated that the plant systems are designed to perform their function in case of a seismic event. For those electrical components which have not been previously evaluated for seismic and environmental qualification the licensee has agreed to prepare a list of these components and a review of their seismic and environmental qualification. The staff will review the qualification status of this equipment upon submittal of this information. In as much as the equipment required for ECCS inside containment has been designed to function in the LOCA environment, and the equipment outside containment necessary for ECCS is not expected to encounter severe environmental effects during a LOCA, we conclude that for this refueling cycle there is reasonable assurance that equipment required for ECCS in event of a LOCA will perform its function.

2.4.2 Steamline Break Accident

The steamline break accident analysis as presented by the licensee is acceptable based on our generic review of NEDO-20360[3].

2.4.3 Fuel Loading Error

Fuel loading errors are discussed in Reference 1 for an 8 x 8 fuel bundle placed in an improper location or rotated 180 degrees in a location near the center of the core. The information in

Reference 1 indicates that a fuel loading error results in a peak linear heat generation rate (LHGR) of 15.95 kW/ft and a minimum critical power ratio (MCPR) of 1.05 in the misplaced 8 x 8 (2.62% enrichment) fuel bundle during steady state operation. The peak LHGR is less than that required to cause a 1% plastic strain in the cladding. Although the MCPR for the bundle is lower than the core wide safety limit of 1.06, it is sufficiently above 1.00 to achieve a high probability of avoiding boiling transition. Fuel bundles adjacent to a misplaced fuel bundle will be negligibly affected. We conclude that the consequences of a fuel loading error are acceptable.

2.4.4 Control Rod Drop Accident

The control rod drop accident for the Dresden Unit 2 re-loaded core is within the bounding analysis presented in Reference 3. The Doppler coefficient of reactivity, the accident reactivity shape and magnitude function, and the rod drop scram reactivity functions are compared with the technical bases presented in Reference 3. This analysis is performed for Doppler coefficients of reactivity at the beginning of Cycle 5, zero void fraction, and at both cold (20°C) and hot (286°C) startup conditions. Incremental bank withdrawal is also assumed. It is shown by Figures 6-1, 6-2, 6-3, 5-4 and 6-5 of Reference 1 that the maximum values of the parameters for this reloaded core will not exceed the bounding values. Therefore, we conclude that the consequences of a control rod drop accident from any insequence control rod during startup will be below the design limit of 280 cal/gm.

2.4.6 Fuel Handling Accident

With respect to fuel handling accidents, in Reference 1, the applicant noted that the general conclusions reached in the generic 8 x 8 reload report (Reference 3) are applicable to this reload: i.e., The total activity released to the environment and the radiological exposures for the 8 x 8 fuel will be less than those values presented in the FSAR for the 7 x 7 core. As identified in the FSAR the radiological exposures for this accident with 7 x 7 fuel are well below the guidelines set forth in 10 CFR 100. Therefore, it is concluded that the consequences of this accident for the 8 x 8 fuel will also be well below the 10 CFR 100 guidelines.

2.5 Overpressure Analysis

Reference 13 states that the overpressure analysis of the lead plant (Quad Cities Unit 2) as presented in Reference 14 is applicable to Dresden Unit 2. That analysis demonstrates that an adequate margin exists below the ASME code allowable vessel pressure of 110% of vessel design pressure. The transient analyzed was the closure of all main steam isolation valves with high neutron flux scram. The analysis was performed for 100% power with the end of cycle scram reactivity insertion rate curve, scram initiated by high neutron flux, void reactivity applicable to this reload, no credit for relief function of safety/relief valves, and one safety valve fails to operate. This analysis utilized input parameters which were equal to or more severe than those which will be experienced during this fuel cycle. The results of the analysis indicate that the peak pressure at the vessel bottom was calculated to be 1327 psig yielding a 48 psi margin below the code allowable, which is acceptable to the staff.

3.0 TECHNICAL SPECIFICATION CHANGES

The proposed Technical Specification changes based on GETAB for Dresden Unit 2 identify the same Fuel Cladding Integrity Safety Limit MCPR of 1.06, but different operating limit MCPR's for the fuel types. We accept the incorporation of the Operating Limit MCPR's of Reference 1 into the Technical Specification for the Dresden Unit 2.

The proposed Technical Specification Limiting Conditions of Operation present two limitations on power distribution related to the LOCA analysis. These are the limiting assembly maximum average planar power density, MAPLHGR, and the minimum power ratio limit related to boiling crisis, MCPR. The MCPR value used in the LOCA analysis was 1.18 and this value is less than the value determined from the transient analysis which will be incorporated in the proposed Technical Specifications. The bases for establishing the limiting value of MAPLHGR are indicated in Section 3.5.I.

The licensee did not include the equalizer line area in the LOCA analysis, therefore, the Technical Specifications will require that the equalizer line valves remain closed at all times during reactor operation. The LOCA analysis did not address one loop operation, therefore, the Technical Specifications will continue to prohibit continuous operation with one loop out of service. The reactor may operate for periods up to 24 hours with one recirculation loop out of service. This short period of time permits corrective action to be taken and reduces the number of unnecessary shutdowns which is consistent with other Technical Specifications. During this period the reactor will be operated within the restrictions of the thermal analysis and will be protected from fuel damage resulting from anticipated transients.

The LOCA analysis assumed all ADS valves operated for small line breaks with HPCI failure. Since the licensee did not provide a LOCA analysis with one ADS valve out of service for small line breaks, the Technical Specifications will require all ADS valves must be operable during reactor operation except that one valve may be out of service for up to seven days if the HPCI is tested daily.

4.0 ENVIRONMENTAL CONSIDERATIONS

The Commission's staff has evaluated the potential for environmental impact associated with operation of Dresden Unit No. 2 in the proposed manner. From this evaluation, the staff had determined that there will be no change in effluent types or total amounts, no change in authorized power level and no significant environmental impact attributable to the proposed action. Having made this determination, the Commission has further concluded pursuant to 10 CFR Section 51.5(c)(1) that no environmental impact statement need be prepared for this action. A Negative Declaration and supporting Environmental Impact Appraisal are being issued with this amendment to the license. As required by Part 51, the Negative Declaration is being filed with the Office of the Federal Register for publication.

5.0 CONCLUSION

Based on our evaluation of reactor operation with Reload-2 fuel, we have concluded that because this change does not involve a significant increase in the probability or consequences of accident 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. Based on our evaluation of operating limits based upon GETAB and on an acceptable ECCS evaluation model, 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. We have also 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 that the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: May 23, 1976

REFERENCES

1. "General Electric BWR Reload-2 Licensing Submittal for Dresden Unit 2 Nuclear Power Station" NEDO-21145 75NED75 Class I, December 1975.

2. Commonwealth Edison Letter (Abrell) to NRC (Ziemann) "Dresden Station Unit 2, License No. DPR-19 Proposed Technical Specification Change for Reload 2 (Cycle 5), NRC Docket No. 50-237", dated April 26, 1976.

3. "General Electric BWR Generic Reload Application for 8 x 8 Fuel" NEDO-20360 Supplement 2 to Revision 1, dated May 30, 1975.

4. Commonwealth Edison Letter (Bolger) to NRC (Rusche), "Dresden Unit 2 Proposed Amendment to Facility Operating License DPR-19, and Appendix A Technical Specifications, NRC Docket No. 50-237," dated March 15, 1976.

5. Commonwealth Edison Letter (Abel) to NRC (Skovholt), "Dresden Station Special Report No. 29, Supplement B - Dresden Station Unit 3 Transient Analyses for Cycle 3 and Quad-Cities Unit 1 Cycle 2, AEC Dockets 50-249 and 50-254," dated March 29, 1974.

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

7. "General Electric BWR Thermal Basis (GETAB): Data, Correlation and Design Application," "NEDO - 10958, 73NED9, Class I, November 1973.

8. General Electric letter (Hinds) to AEC (Butler) "Responses to the Third Set of AEC Questions on the General Electric Licensing Topical Reports, NEDO - 10558 and NEDO 10958, "General Electric BWR Thermal Analysis Basis (GETAB): Data Correlation and Design Application," dated July 24, 1974.

9. Commonwealth Edison letter (Abel) to NRC (Rusche) "Dresden Station Unit 2 Proposed Amendment to Facility Operating Licenses DPR-19 NRC Docket 50-237" dated September 3, 1975.

10. Commonwealth Edison letter (Bolger) to NRC (Rusche) "Dresdent Station Unit 2 Proposed Amendment to Facility Operating License DPR-19 and Dresden Station Special Report No. 40, Supplement B, NRC Docket No. 50-237", dated July 1, 1975.

11. Commonwealth Edison letter (Abel) to NRC (Ziemann) "Quad-Cities Station Unit 2, Special Report No. 15, Supplement C, NRC Docket No. 50-265", dated April 8, 1975.

12. Commonwealth Edison letter (Abrell) to NRC (Ziemann) "Quad-Cities Station Unit 2 Special Report No. 15 Supplement C NRC Docket No. 50-254 and 50-265, dated February 26, 1976.

13. Commonwealth Edison letter (Abrell) to NRC (Ziemann) "Dreden Station Unit 2, License No. DPR-19 Proposed Technical Specification change for Reload 2 (Cycle 5), NRC Docket No. 50-237", dated April 28, 1976.

14. Commonwealth Edison letter (Abel) to NRC (Ziemann) "Quad-Cities Station Unit 2 Reload No. 1 Licensing Submittal Supplement E NRC Docket No. 50-265", dated April 16, 1975.

15. Letter: G. A. Abrell, Commonwealth Edison to D. L. Ziemann, NRC
Subject: Quad Cities Special Report No. 15 Supplement C
Date: February 26, 1976

16. Letter: R. L. Bolger, Commonwealth Edison to D. L. Ziemann, NRC
Subject: Dresden Station Unit 2
Date: April 6, 1976

17. Letter: J. S. Abel, Commonwealth Edison to D. L. Ziemann, NRC
Subject: Dresden Station Units 2 and 3 Quad Cities Station Units 1 and 2 DresdenStation Special Report No. 40 Quad Cities Station Special Report No. 15
Date: July 10, 1975

18. Letter: J. S. Abel, Commonwealth Edison to D. L. Ziemann, NRC
Subject: Dresden Station Units 2 and 3, Quad Cities Station Units 1 and 2, Dresden Station Special Report No. 40, Quad Cities Station Special Report No. 15
Date: July 7, 1975

19. Letter: J. S. Abel, Commonwealth Edison to D. L. Ziemann, NRC
Subject: Dresden Station Units 2 and 3, Quad Cities Station Units 1 and 2, Undesirable Function of Motor Operated Valves
Date: June 23, 1975

20. Letter: G. A. Abrell, Commonwealth Edison to D. L. Ziemann, NRC
Subject: Dresden Station Unit 2 ECCS Appendix K Single Failure Analysis
Date: May 17, 1976

21. Letter: R. L. Bolger, Commonwealth Edison to D. L. Ziemann, NRC
Subject: ECCS Single Failure Review
Date: May 21, 1976

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-237

COMMONWEALTH EDISON COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO PROVISIONAL OPERATING LICENSE

Notice is hereby given that the U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 21 to Provisional Operating License No. DPR-19 to the Commonwealth Edison Company (the licensee), which revised the license and its appended Technical Specifications for operation of the Dresden Nuclear Power Station Unit No. 2 (the facility) located in Grundy County, Illinois. The amendment is effective as of its date of issuance.

The amendment revised the provisions in the license and its Technical Specifications for the facility to authorize operation (1) with additional 8 x 8 uranium 235 fuel assemblies, and (2) using modified operating limits based on an acceptable evaluation model that conforms with Section 50.46 of 10 CFR Part 50, and with operating limits based on the General Electric Thermal Analysis Basis (GETAB), in accordance with the licensee's applications for the amendment as referenced in the last paragraph of this notice.

The applications for the amendment 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 amendment. Notice of Proposed Issuance of Amendment to Provisional Operating License in connection with item (2) above was published in the FEDERAL REGISTER on December 2, 1975 (40 FR 55908). No request for a hearing or petition for leave to intervene was filed following notice of the proposed action on item (2) above. Prior public notice of item (1) above was not required since the action does not involve a significant hazards consideration. In connection with the action on Section 50.46 regarding emergency core cooling system (part of item 2) the Commission has issued a Negative Declaration and Environmental Impact Appraisal. In connection with the action identified as item (1) of this Notice, the Commission has determined that the action will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental statement, negative declaration or environmental impact appraisal need not be prepared.

For further details with respect to this action, see (1) the applications for amendment dated July 1, 1975, September 3, 1975, March 15, 1976, and supplements dated July 7 and 10, August 25, September 19, 1975, February 26, 1976, April 6, 9, 19, 26 and 28, and May 17 and 21, 1976, (2) the April 8, 1975 Quad Cities Unit No. 2 licensee submittal in Docket No. 50-265 which is applicable to Dresden 2 and is the non-proprietary version of the Dresden 2 proprietary submittal dated July 21, 1975, (3) Amendment No. 21 to License No. DPR-19, (4) the Commission's concurrently issued related

Safety Evaluation, and (5) the Commission's Negative Declaration dated (which is also being published in the FEDERAL REGISTER) and associated Environmental Impact Appraisal. 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 Morris Public Library at 604 Liberty Street in Morris, Illinois 60451. A single copy of items (2) through (5) 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 23rd day of May, 1976.

FOR THE NUCLEAR REGULATORY COMMISSION

Dennis L. Ziemann

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