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Docket No. 50-237
LS05-81-03-081

MARCH 31 1981

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

Dear Mr. Abel:

In response to your application dated January 28, 1981, supplemented by your letter dated February 23, 1981, the Commission has issued the enclosed Amendment No. 58 to Provisional Operating License No. DPR-19 for Dresden Nuclear Power Station Unit 2.

This amendment (1) authorizes changes to the plant Technical Specifications which you proposed to support your review of future reloads for Dresden Unit 2, under provisions of 10 CFR 50.59 and (2) modifies license condition 3.F to assure a conservative MCPR operating limit during coastdown operation.

During our review, changes were made to your submittal. These have been discussed with and agreed to by your staff.

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

Sincerely,
Original signed by
Dennis M. Crutchfield

Dennis M. Crutchfield, Chief
Operating Reactors Branch #5
Division of Licensing

Enclosures:
1. Amendment No. 58 to DPR-19
2. Safety Evaluation
3. Notice

cc w/enclosures:
See next page

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SURNAME	HSmith:dn	PO'Connor	JOlshinski		DCrutchfield	GCLainas
DATE	3/3/81	3/3/81	3/ /81	3/21/81	3/31/81	3/31/81

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

March 31, 1981

Docket No. 50-237
LS05-81-03-081

Mr. J.S. Abel
Director of Nuclear Licensing
Commonwealth Edison Company
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Operating Reactors Branch #5
Division of Licensing

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2. Safety Evaluation
3. Notice

cc w/enclosures:
See next page

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-237

DRESDEN NUCLEAR POWER STATION UNIT NO. 2

AMENDMENT TO PROVISIONAL OPERATING LICENSE

Amendment No. 58
License No. DPR-19

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

   A. The application for amendment by the Commonwealth Edison Company (the licensee) dated January 28, 1981, supplemented by your letter dated February 23, 1981, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

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2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 3.B and 3.F of Provisional Operating License No. DPR-19 are hereby amended to read as follows:

   3.B Technical Specifications

   The Technical Specifications contained in Appendix A as revised through Amendment No. 58 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

   3.F Restrictions

   Operation in the coastdown mode is permitted to 40% power. Should off-normal feedwater heating be necessary for extended periods during coastdown (i.e., greater than 24 hours) the licensee shall perform a safety evaluation to determine if the MCPR Operating Limit and calculated peak pressure for the worst case abnormal operating transient remain bounding for the new condition.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Dennis M. Crutchfield

Dennis M. Crutchfield, Chief
Operating Reactors Branch #5
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 31, 1981

ATTACHMENT TO LICENSE AMENDMENT NO. 58

PROVISIONAL OPERATING LICENSE NO. DPR-19

DOCKET NO. 50-237

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages.

2	81
4	81B
5	81B-1
6	81C-1
7	81C-2
9 (Blank page)	81D
10	82
11	85A
12	85B
13	86A
14	90
15	
16	
18	
20	
21	
22	Add pages: 81C-3
26	81C-4 and
29	81C-5
34	
42	
42A	
45	
46	
47	
48	
49	
57	
57A	
60	
62A	
63	
64	
71	

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. Fraction of Limiting Power Density (FLPD) - The fraction of limiting power density is the ratio of the Linear Heat Generation Rate (LHGR) existing at a given location to the design LHGR for that bundle type.

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.

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 conbined interval time for any 3 consecutive intervals not to exceed 3.25 times the specified surveillance interval.

DD. Fraction of Rated Power (FRP) - The fraction of rated power is the ratio of core thermal power to rated thermal power of 2527 Mwth.

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.

FF. Maximum Fraction of Limiting Power Density (MFLPD) - The maximum fraction of limiting power density is the highest value existing in the core of the Fraction of Limiting Power Density (FLPD).

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 1.07 shall constitute violation of the MCPR 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:

$$S \leq [.65W_D + 55]$$

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

In the event of operation with a maximum fraction of limiting power density (MFLPD) greater than the fraction of rated power (FRP), the setting shall be modified as follows:

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

Where:

FRP = fraction of rated thermal power (2527 MWt)

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

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

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

2. APRM Flux Scram Trip Setting (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

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

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

C. Power Transient

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

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

D. Reactor Water Level (Shutdown Condition)

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

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

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:

$$S \leq \left[.65W_D + 43\right]$$

The definitions used above for the APRM scram trip apply.

In the event of operation with a maximum fraction limiting power density (MFLPD) greater than the fraction of rated power (FRP), the setting shall be modified as follows:

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

The definitions used above for the APRM scram trip apply.

The ratio of FRP to MFLPD shall be set equal to 1.0 unless the actual operating value is less than 1.0, in which case the actual operating value will be used. This adjustment may also be performed by increasing the APRM gain by the inverse ratio, MFLPD/FRP, which accomplishes the same degree of protection as reducing the trip setting by FRP/MFLPD.

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

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

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

Onset of transition boiling results in a decrease in heat transfer from the clad and, therefore, elevated clad temperature and the possibility of clad failure. However, the existence of critical power, or boiling transition, is not a directly observable parameter in an operating reactor. Therefore, the margin to boiling transition is calculated from plant operating parameters such as core power, core flow, feedwater temperature, and core power distribution. The margin for each fuel assembly is characterized by the critical power ratio (CPR) which is the ratio of the bundle power which would produce onset of transition

Safety Limit Bases

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

boiling divided by the actual bundle power. The minimum value of this ratio for any bundle in the core is the minimum critical power ratio (MCPR). It is assumed that the plant operation is controlled to the nominal protective setpoints via the instrumented variables. (Figure 2.1-3).

The MCPR fuel cladding integrity safety limit 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, is derived from a detailed statistical analysis considering all of the uncertainties in monitoring the core operating state including uncertainty in the boiling transition correlation. See e. g. Reference (1).

Because the boiling transition correlation is based on a large quantity of full scale data there is a very high confidence that operation of a fuel assembly at the condition of MCPR = the MCPR fuel cladding integrity safety limit 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 all 8x8 fuel types. This constraint is established by Specification 3.5.J to provide adequate safety margin to 1% plastic strain for abnormal operating transients initiated from high power conditions. Specification 2.1.A.1 provides for equivalent safety margin for transients initiated from lower power conditions by adjusting the APRM flow biased scram by the ratio of FRP/MFLPD.. Specification 3.5.J established the LHGR max which cannot be exceeded under steady power operation.

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

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

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.

Control rod scram times are checked as required by Specification 4.3.C. Exceeding a neutron flux scram setting and a failure of the control rods to reduce flux to less than the scram setting within 1.5 seconds does not necessarily imply that fuel is damaged; however, for this specification a safety limit violation will be assumed any time a neutron flux scram setting is exceeded for longer than 1.5 seconds.

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

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

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 2527 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.

Conservatism incorporated into the transient analyses is documented in Reference 1. Transient analyses are initiated at the conditions given in this reference.

Amendment No. 58

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 20% 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.

DPR-19

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

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

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

A. Neutron Flux Trip Settings

1. APRM Flux Scram Trip Setting (Run Mode)

The average power range monitoring (APRM) system, which is calibrated using heat balance data taken during steady-state conditions, reads in percent of rated thermal power. Because fission chambers provide the basic input signals, the APRM system responds directly to average neutron flux. During transients, the instantaneous rate of heat transfer from the fuel (reactor thermal power) is less than the instantaneous neutron flux due to the time constant of the fuel. Therefore, during abnormal operational transients, the thermal power of the fuel will be less than that indicated by the neutron flux at the scram setting. Analyses demonstrate that with a 120 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.

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 maximum Fraction of Limiting Power Density (MFLPD) and reactor core thermal power. The scram setting is adjusted in accordance with the formula in Specification 2.1.A.1, when the MFPLD is greater than the Fraction of Rated Power (FRP). The adjustment may be accomplished by increasing the APRM gain by the reciprocal of FRP/MFLPD. This provides the same degree of protection as reducing the trip setting by FRP/MFLPD by raising the initial APRM reading closer to the trip setting such that a scram would be received at the same point in a transient as if the trip setting had been reduced.

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 [illegible] 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) 2.1.B

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 the MCPR fuel cladding integrity safety limit 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.

APRM Rod Block Trip Setting

Reactor power level may be varied by moving control rods or by varying the recirculation flow rate. The APRM system provides a control rod block to prevent gross rod withdrawal at constant recirculation flow rate to protect against grossly exceeding the MCPR fuel cladding integrity safety limit. This rod block trip setting, which is automatically varied with recirculation loop flow rate, prevents an increase in the reactor power level to excessive values due to control rod withdrawal. The flow variable trip setting provides substantial margin from fuel damage, assuming a steady-state operation at the trip setting, over the entire recirculation flow range. The margin to the Safety Limit increases as the flow decreases for the specified trip setting versus flow relationship; therefore the worst case MCPR which could occur during steady-state operation is at 108% of rated thermal power because of the APRM rod block trip setting. The actual power distribution in the core is established by specified control rod sequences and is monitored continuously by the in-core LPRM system. As with the APRM scram trip setting, the APRM rod block trip setting is adjusted downward if the maximum fraction of limiting power density exceeds the fraction of rated power, thus preserving the APRM rod block safety margin.

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 the MCPR fuel cladding integrity safety limit, 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 the MCPR fuel cladding integrity safety limit for this transient. For the load rejection without bypass transient from 100% power, the peak heat flux (and therefore LHGR) increases on the order of 15% which provides wide margin to the value corresponding to 1% plastic strain of the cladding.

G. Reactor Coolant Low Pressure Initiates Main Steam Isolation Valve Closure - The low pressure isolation at 850 psig was provided to give protection against fast reactor depressurization and the resulting rapid cooldown of the vessel. Advantage was taken of the scram feature which occurs 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.

Bases:

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

The pressure safety limit of 1325 psig as measured by the vessel steam space pressure indicator is equivalent to 1375 psig at the lowest elevation of the reactor coolant system. The 1375 psig value is derived from the design pressures of the reactor pressure vessel, coolant system piping and isolation condenser. The respective design pressures are 1250 psig at 575°F, 1175 psig at 560°F, and 1250 psig at 575°F. The pressure safety limit was chosen as the lower of the pressure transients permitted by the applicable design codes: ASME Boiler and Pressure Vessel Code, Section III for the pressure vessel and isolation condenser and USASI B31.1 Code for the reactor coolant system piping. The ASME Boiler and Pressure Vessel Code permits pressure transients up to 10% over design pressure (110% X 1250 = 1375 psig), and the USASI Code permits pressure transients up to 20% over the design pressure (120% X 1175 = 1410 psig). The Safety Limit pressure of 1375 psig is referenced to the lowest elevation of the primary coolant system. Evaluation methodology used to assure that this safety limit pressure is not exceeded for any period is documented in Reference 1. The design basis for the reactor pressure vessel makes evident the substantial margin of protection against failure at the safety pressure limit of 1375 psig. The vessel has been designed for a general membrane stress no greater than 26,700 psi at an internal pressure of 1250 psig; this is a factor of 1.5 below the yield strength of 40,100 psi at 575°F. At the pressure limit of 1375 psig, the general membrane stress will only be 29,400 psi, still safely below the yield strength.

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

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

Finally, the safety valves are sized to keep the reactor coolant system pressure below 1375 psig with no credit taken for the relief valves during the postulated full closure of all MSIV's without direct (valve position switch) scram. Credit is taken for the neutron flux scram, however the indirect flux scram and safety valve actuation provide adequate margin below the peak allowable vessel pressure of 1375 psig.

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

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(4) SAR, Section 11.2.2.

Bases:

2.2 In compliance with Section III of the ASME Code, the safety valves must be set to open at no higher than 103% of design pressure, and they must limit the reactor pressure to no more than 110% of design pressure. Both the neutron flux scram and safety valve actuation are required to prevent overpressurizing the reactor pressure vessel and thus exceeding the pressure safety limit. The pressure scram is available as a backup protection to the high flux scram.

If the high flux scram were to fail, a high pressure scram would occur at 1060 psig. Analyses are performed as described in the Generic Reload Fuel Application, NEDE-24011-P-A (Approved revision number at time reload analyses are performed) for each reload to assure that the pressure safety limit is not exceeded.

TABLE 4.1.1 (cont)

Notes:

1. Initially once per month until exposure hours (M as defined on Figure 4.1.1) is 2.0×10^5; thereafter, according to Figure 4.1.1 with an interval not less than one month nor more than three months. The compilation of instrument failure rate data may include data obtained from other Boiling Water Reactors for which the same design instrument operates in an environment similar to that of Dresden Unit 2.

2. An instrument check shall be performed on low reactor water level once per day and on high steamline radiation once per shift.

3. A description of the three groups is included in the bases of this Specification.

4. Functional tests are not required when the systems are not required to be operable or are tripped. If tests are missed, they shall be performed prior to returning the systems to an operable status.

5. This instrumentation is exempted from the Instrument Functional Test Definition (1.F). This instrument Functional Test will consist of injecting a simulated electrical signal into the measurement channels.

* 6. If reactor start-ups occur more frequently than once per week, the functional test need not be performed; i.e., the maximum functional test frequency shall be once per week.

The control rod drive scram system is designed so that all of the water which is discharged from the reactor by a scram can be accommodated in the discharge piping. A part of this piping is an instrument volume (u-tube in the piping) which accommodates in excess of 50 gallons of water and is the low point in the piping. No credit was taken for this volume in the design of the discharge piping as concerns the amount of water which must be accommodated during a scram. During normal operation the discharge volume is empty; however, should it fill with water, the water discharged to the piping from the reactor could not be accommodated which would result in slow scram times or partial or no control rod insertion. To preclude this occurrence, level switches have been provided in the instrument volume which alarm and scram the reactor when the volume of water reaches 50 gallons. As indicated above, there is sufficient volume in the piping to accommodate the scram without impairment of the scram times or amount of insertion of the control rods. This function shuts the reactor down while sufficient volume remains to accommodate the discharged water and precludes the situation in which a scram would be required but not be able to perform its function adequately.

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

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

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

A reactor mode switch is provided which actuates or bypasses the various scram functions appropriate to the particular plant operating status. Ref. Section 7.7.1.2 SAR.

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

The IRM system provides protection against excessive power levels and short reactor periods in the start-up and intermediate power ranges. Ref.

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 MFLPD 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 MFLPD is adequate.

INSTRUMENTATION THAT INITIATES ROD BLOCK

Table 3.2.3

Minimum No. of Operable Inst. Channels Per Trip System(1)	Instrument	Trip Level Setting
1	APRM upscale (flow bias) (7)	$\leq [.65W_D + 43]$ FRP MFPLD (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)	$\leq [.65W + 42]$ (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	(4)
2(5)	SRM detector not in startup position	
2(5)(6)	SRM upscale	$\leq 10^5$ counts/sec

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), and RBM downscale need not be operable in the Startup/Hot Standby mode. The RBM upscale need not be operable at less than 30% rated thermal power. One channel may be bypassed above 30% rated thermal power provided that a limiting control rod pattern does not exist. For systems with more than one channel per trip system, If the first column cannot be met for both trip systems, the systems shall be tripped.

2. W_D percent of drive flow required to produce a rated core flow of 98 Mlb/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 durii or after refueling at power levels not to exceed 5 MW(t).

TABLE 4.2.1 (cont)

Notes:

1. Initially once per month until exposure hours (M as defined on Figure 4.1.1) is 2.0×10^5; thereafter, according to Figure 4.1.1 with an interval not less than one month nor more than three months. The compilation of instrument failure rate data may include data obtained from other Boiling Water Reactors for which the same design instrument operates in an environment similar to that of Dresden Unit 2.

2. Functional test calibrations and instrument checks are not required when these instruments are not required to be operable or are tripped. Functional tests shall be performed before each startup with a required frequency not to exceed once per week. Calibrations shall be performed during each startup or during controlled shutdowns with a required frequency not to exceed once per week. Instrument checks shall be performed at least once per week. Instrument checks shall be performed at least once per day during those periods when the instruments are required to be operable.

3. This instrumentation is excepted from the functional test definition. The functional test will consist of injecting a simulated electrical signal into the measurement channel. See Note 4.

4. These instrument channels will be calibrated using simulated electrical signals once every three months. In addition, calibration including the sensor will be performed during each refueling outage.

5. A minimum of two channels is required.

6. From and after the date that one of these parameters [... either drywell-torus differential pressure or torus water level indication] is reduced to one indication, continued operation is not permissible beyond thirty days, unless such instrumentation is sooner made operable. In the event that all indications of these parameters [...either drywell-torus differential pressure or torus water level] is disabled and such indication cannot be restored in six (6) hours, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition in twenty four hours.

Bases:

3.2 In addition to reactor protection instrumentation which initiates a reactor scram, protective instrumentation has been provided which initiates action to mitigate the consequences of accidents which are beyond the operators ability to control, or terminates operator errors before they result in serious consequences. This set of Specifications provides the limiting conditions of operation for the primary system isolation function, initiation of the emergency core cooling system, control rod block and standby gas treatment systems. The objectives of the specifications are (i) to assure the effectiveness of the protective instrumentation when required by preserving its capability to tolerate a single failure of any component of such systems even during periods when portions of such systems are out of service for maintenance, and (ii) to prescribe the trip settings required to assure adequate performance. When necessary, one channel may be made inoperable for brief intervals to conduct required functional tests and calibrations.

Some of the settings on the instrumentation that initiates or controls core and containment cooling have tolerances explicitly stated where the high and low values are both critical and may have a substantial effect on safety. It should be noted that the setpoints of other instrumentation, where only the high or low end of the setting has a direct bearing on safety, are chosen at a level away from the normal operating range to prevent inadvertent actuation of the safety system involved and exposure to abnormal situations.

Isolation valves are installed in those lines that penetrate the primary containment and must be isolated during a loss of coolant accident so that the radiation dose limits are not exceeded during an accident condition. Actuation of these valves is initiated by protective instrumentation shown in Table 3.2.1 which senses the conditions for which isolation is required. Such instrumentation must be available whenever primary containment integrity is required. The objective is to isolate the primary containment so that the guidelines of 10 CFR 100 are not exceeded during an accident.

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

The low-reactor water level instrumentation is set to trip at >8 inches on the level instrument (top of active fuel is defined to be 360 inches above vessel zero) and after allowing for the full power pressure drop across the steam dryer the low level trip is at 504 inches above vessel zero, or 144 inches above the top of active fuel. Retrofit 8x8 fuel has an active fuel length 1.24 inches longer than earlier fuel designs. However, present trip setpoints were used in the LOCA analyses (NEDO-24146A, April 1979). This trip initiates closure of Group 2 and 3 primary containment isolation valves but does not trip the recirculation pumps (reference SAR Section 7.7.2). For a trip setting of 504 inches above vessel zero (144 inches above top of active fuel) and a 60-second valve closure time, the valves will be closed before perforation of the cladding occurs even for the maximum break; the setting is therefore adequate.

The low low reactor level instrumentation is set to trip when reactor water level is 444 inches above vessel zero (with top of active fuel defined as 360 inches above vessel zero, -59 inches is 84 inches above the top of active fuel).

This trip initiates closure of Group 1 primary containment isolation valves, Ref. Section 7.7.2.2 SAR, and also activates the ECC subsystems, starts the emergency diesel generator and trips the recirculation pumps. This trip setting level was chosen to be high enough to prevent spurious operation but low enough to initiate ECCS operation and primary system isolation so that no melting of the fuel cladding will occur and so that post accident cooling can be accomplished and the guidelines of 10 CFR 100 will not be violated. For the complete circumferential break of a 28-inch recirculation line and with the trip setting given above, ECCS initiation and primary system isolation are initiated in time to meet the above criteria.

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

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

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

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

High radiation monitors in the main steamline tunnel have been provided to detect gross fuel failure. This instrumentation causes closure of Group 1 valves, the only valves required to close for this accident. With the established setting of 3 times normal background, and main steamline isolation valve closure, fission product release is limited so that 10 CFR 100 guidelines are not exceeded for this accident. Ref. Section 14.2.1.7 SAR. The performance of the process radiation monitoring system relative to detecting fuel leakage shall be evaluated during the first five years of operation. The conclusions of this evaluation will be reported to the Atomic Energy Commission.

Pressure instrumentation is provided which trips when main steamline pressure drops below 850 psig. A trip of this instrumentation results in closure of Group 1 isolation valves. In the "Refuel" and "Startup/Hot Standby" mode this trip function is bypassed. This function is provided primarily to provide protection against a pressure regulator malfunction which would cause the control and/or bypass valves to open. With the trip set at 850 psig inventory loss is limited so that fuel is not uncovered and peak clad temperatures are much less than 1500°F; thus, there are no fission products available for release other than those in the reactor water. Ref. Section 14.2.3 SAR.

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 instrumentation 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 go below the MCPR fuel cladding integrity safety limit. 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 expecially during operation at reduced flow. The APRM provides gross core protection; i.e., limits the gross withdrawal of control rods in the normal withdrawal sequence.

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

prevents control rod withdrawal before a scram is reached.

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

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

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 the MCPR fuel cladding integrity safety limit.

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 CONDITION FOR OPERATION

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

(b) Whenever the reactor is in the startup or run mode below 20% rated thermal power, the Rod Worth Minimizer shall be operable. A second operator or qualified technical person may be used as a substitute for an inoperable Rod Worth Minimizer which fails after withdrawal of at least 12 control rods to the fully withdrawn position. The Rod Worth Minimizer may also be bypassed for low power physics testing to demonstrate the shutdown margin requirements of specifications 3.3.A.1 if a nuclear engineer is present and verifies the step-by-step rod movements of the test procedure.

4.3 SURVEILLANCE REQUIREMENTS

3. (a) To consider the rod worth minimizer operable, the following steps must be performed:

(i) The control rod withdrawal sequence for the rod worth minimizer computer shall be verified as correct.

(ii) The rod worth minimizer computer on-line diagnosite test shall be successfully completed.

(iii) Proper annunciation of the select error of at least one out-of-sequence control rod in each fully inserted group shall be verified.

(iv) The rod block function of the rod worth minimizer shall be verified by attempting to withdraw an out-of-sequence control rod beyond the block point.

(b) If the rod worth minizer is inoperable while the reactor is in the startup or run mode below 20% rated thermal power and a second independent operator or engineer is being used, he shall verify that all rod positions are correct prior to commencing withdrawal of each rod group.

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 that the MCPR will remain above the MCPR fuel cladding integrity safety limit assuming a single error that results in complete withdrawal of any single operable control rod.

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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.

3.3 LIMITING CONDITION FOR OPERATION	4.3 SURVEILLANCE REQUIREMENT
E. Reactivity Anomalies The reactivity equivalent of the difference between the actual critical rod configuration and the expected configuration during power operation shall not exceed 1% ΔK. If this limit is exceeded, the reactor will be shutdown until the cause has been determined and corrective actions have been taken if such actions are appropriate. In accordance with Specification 6.6, the NRC shall be notified of this reportable occurrence within 24 hours.	E. Reactivity Anomalies During the startup test program and startups following refueling outages, the critical rod configurations will be compared to the expected configurations at selected operating conditions. These comparisons will be used as base data for reactivity monitoring during subsequent power operation throughout the fuel cycle. At specific power operating conditions, the critical rod configuration will be compared to the configuration expected based upon appropriately corrected past data. This comparison will be made at least every equivalent full power month.
F. If Specifications 3.3.A through D above are not met, an orderly shutdown shall be initiated and the reactor shall be in the Cold Shutdown condition within 24 hours.	
G. Economic Generation Control System Operation of the unit with the Economic Generation Control system with automatic flow control shall be permissible only in the range of 65-100% of rated core flow, with reactor power above 20%.	G. Economic Generation Control System Weekly, the range set into the Economic Generation Control System shall be recorded.

indicative of a generic control rod drive problem and the reactor will be shutdown. Also if damage within the control rod drive mechanism and in particular, cracks in drive internal housings, cannot be ruled out, then a generic problem affecting a number of drives cannot be ruled out. Circumferential cracks resulting from stress assisted intergranular corrosion have occurred in the collet housing of drives at several BWRs. This type of cracking could occur in a number of drives and if the cracks propagated until severance of the collet housing occurred, scram could be prevented in the affected rods. Limiting the period of operation with a potentially severed collet housing and requiring increased surveillance after detecting one stuck rod will assure that the reactor will not be operated with a large number of rods with failed collet housings.

c. Control Rod Withdrawal

1. Control rod dropout accidents as discussed in Reference 6 can lead to significant core damage. If coupling integrity is maintained, the possibility of a rod dropout accident is eliminated. The overtravel position feature provides a positive check as only uncoupled drives may reach this position. Neutron instrumentation response to rod movement provides a verification that the rod is following its drive. Absence of such response to drive movement would provide cause for suspecting a rod to be uncoupled and stuck. Restricting recoupling verifications to power levels above 20% provides assurance that a rod drop during a recoupling verification would not result in a rod drop accident more severe than analyzed.

2. The control rod housing support restricts the outward movement of a control rod to less than 3 inches in the extremely remote event of a housing failure. The amount of reactivity which could be added by this small amount of rod withdrawal, which is less than a normal single withdrawal increment, will not contribute to any damage to the primary coolant system. The design basis is given in Section 6.6.1 of the SAR, and the design evaluation is given in Section 6.6.3. This support is not required if the reactor coolant system is at atmospheric pressure since there would then be no driving force to rapidly eject a drive housing. Additionally, the support is not required if all control rods are fully inserted and if an adequate shutdown margin with one control rod withdrawn has been demonstrated since the reactor would remain subcritical even in the event of complete ejection of the strongest control rod.

3. Control rod withdrawal and insertion sequences are established to assure that the maximum insequence individual control rod or control rod segments which are withdrawn could not be worth enough to cause the rod drop accident design limit of 280 cal/ gram to be exceeded if they were to drop out of the core in the manner defined for the Rod Drop Accident.(3) These sequences are developed prior to initial operation of the unit following any refueling outage and the requirement that an operator follow these sequences is backed up by the operation of the RMS or a second qualified station employee. These sequences are developed to limit reactivity worths of control rods and together with the integral rod velocity limiters and the action of the control rod drive system, limit potential reactivity insertion such that the results of a control rod drop accident will not exceed a maximum fuel energy content of 280 cal/gm. The peak fuel enthalpy of 280 cal/gm is below the energy content at which rapid fuel dispersal and primary system damage have been found to occur based on experimental data as is discussed in Reference 1.

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

Bases (cont'd)

These techniques are described in a topical report (1) and two supplements. (2) (3) In addition, a banked position withdrawal sequence described in Reference (4) has been developed to further reduce incremental rod worths.

By using the analytical models described in those reports coupled with conservative or worst-case input parameters, it has been determined that for power levels less than 20% of rated power, the specified limit (typically 1.3% ΔK) on insequence control rod or control rod segment worths will limit the peak fuel enthalpy to less than 280 cal/gm. Above 20% power even single operator errors cannot result in out-of-sequence control rod worths which are sufficient to reach a peak fuel enthalpy of 280 cal/gm should a postulated control rod drop accident occur.

The following parameters and worst-case bounding assumptions have been utilized in the reload analysis to determine compliance with the 280 cal/gm peak fuel enthalpy. Method and basis for the rod drop accident analyses are documented in Reference 6. Each core reload will be analyzed to show conformance to the limiting parameters.

a. An inter-assembly local power peaking factor (5)

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

c. A beginning-of-life Doppler reactivity feedback.

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

e. The maximum possible rod drop velocity (3.11 ft./sec.)

f. The design accident and scram reactivity shape function.

g. The minimum moderator temperature to reach criticality.

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(1) Paone, C.J., Stirn, R.C. and Wooley, J.A., "Rod Drop Accident Analysis for Large Boiling Water Reactors", NEDO-10527, March 1972.

(2) Stirn, R.C., Paone, C.J., and Young, R.M., "Rod Drop Accident Analysis for Large BWR's", Supplement 1 - NEDO-10527, July 1972

(3) Stirn, R.C., Paone, C.J., and Haun, J.M., "Rod Drop Accident Analysis for Large BWR's Addendum No. 2, Exposed Cores", Supplement 2-NEDO 10527, January 1973.

(4) C.J. Paone, "Banked Position Withdrawal Sequence" Licensing Topical Report NEDO-2123, January 1977.

(5) To include the power spike effect caused by gaps between fuel pellets.

(6) Generic Reload Fuel Application NEDE-24011-P-A, August 1978*

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

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 the MCPR fuel cladding integrity safety limit. 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 the MCPR fuel cladding integrity safety limit. Analysis of the limiting power transient shows that the negative reactivity rates resulting from the scram with the average response of all the drives as given in the above specification, provide the required protection, and MCPR remains greater than the MCPR fuel cladding integrity safety limit.

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, 290 milliseconds are allowed between a neutron sensor reaching the scram point and the start of motion of the control rods. This is adequate and conservative when compared to the typically observed time delay of about 210 milliseconds.

Approximately 90 milliseconds after neutron flux reaches the trip point, the pilot scram valve solenoid de-engerizes and 120 milliseconds later the control rod motion is estimated to actually begin. However 200 milliseconds rather than 120 milliseconds is conservatively assumed for this time interval in the transient analyses, and is also included in the allowable scram insertion times specified in Specification 3.3.C.

The scram times for all control rods will be determined at the time of each refueling outage.

Fifty percent of the control rods will be checked every 16 weeks to verify the performance.

The history of drive performance accumulated to date indicates that the 90% insertion times of new and overhauled drives approximate a normal distribution about the mean which tends to become skewed toward longer scram times as operating time is accumulated. The probability of a drive not exceeding the mean 90% insertion time by 0.75 seconds is greater than 0.999 for a normal distribution. The measurement of the scram performance of the drives surrounding a drive exceeding the expected range of

Bases:

A. The design objective of the standby liquid control system is to provide the capability of bringing the reactor from full power to a cold, xenon-free shutdown assuming that none of the withdrawn control rods can be inserted. To meet this objective, the liquid control system is designed to inject a quantity of boron which produces a concentration of no less than 600 ppm of boron in the reactor core in less than 100 minutes. 600 ppm boron concentration in the reactor core is required to bring the reactor from full power to a 3%Δk or more subcritical condition considering the hot to cold reactivity swing, xenon poisoning and an additional margin (25%) for possible imperfect mixing of the chemical solution in the reactor water. A minimum quantity of 3478 gallons of solution having a 13.4% sodium pentaborate concentration is required to meet this shutdown requirement.

The time requirement (100 minutes) for insertion of the boron solution was selected to override the rate of reactivity insertion due to cooldown of the reactor following the xenon poison peak. For a required pumping rate of 39 gallons per minute, the maximum storage volume of the boron solution is established as 4,059 gallons (158 gallons are contained below the pump suction and, therefore, cannot be inserted).

Boron concentration, solution temperature, and volume are checked on a frequency to assure a high reliability of operation of the system should it ever be required. Experience with pump operability indicates that monthly testing is adequate to detect if failures have occurred. Components of the system are checked periodically as described above and make a functional test of the entire system on a frequency of less than once during each operating cycle unnecessary. A test of one installed explosive charge is made at least once during each operating cycle to assure that the charges have not deteriorated, the actuation circuit is functioning properly, the valve functions properly, and no flow blockages exist. The replacement charge will be selected from a batch for which there has been a successful test firing. Recommendations of the vendor shall be followed in maintaining a five-year life of the explosive charges. A continual check of the firing circuit continuity is provided by pilot lights in the control room.

The relief valves in the standby liquid control system protect the system piping and positive displacement pumps which are nominally designed for 1500 psig protection from overpressure. The pressure relief valves discharge back to the standby liquid control solution tank.

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

C. The solution saturation temperature of 13% sodium pentaborate, by weight, is 59°F. To guard against boron precipitation, the solution including that in the pump suction piping is kept at least 10°F above the saturation temperature by a tank heater and by heat tracing in the pump suction piping. The 10°F margin is

3.5 LIMITING CONDITION FOR OPERATION

and control rod drive maintenance performed provided that the spent fuel pool gates are open, the fuel pool water level is maintained above the low level alarm point, and the minimum total condensate storage reserve is maintained at 230,000 gallons, and provided that not more than one control rod drive housing is open at one time, the control rod drive housing is blanked following removal of the control rod drive, no work is being performed in the reactor vessel while the housing is open and a special flange is available which can be used to blank an open housing in the event of a leak.

5. When irradiated fuel is in the reactor and the vessel head is removed, work that has the potential for draining the vessel may be performed with less than 112,000 ft^3 of water in the suppression pool, provided that: 1) the total volume of water in the suppression pool, dryer separator above the shield blocks, refueling cavity, and the fuel storage pool above the bottom of the fuel pool gate is greater than 112,000 ft^3; 2) the fuel storage pool gate is removed; 3) the low pressure coolant injection and core spray systems are operable; and 4) the automatic mode of the drywell sump pumps is disabled.

H. Maintenance of Filled Discharge Pipe

Whenever core spray, LPCI, or HPCI ECCS are required to be operable, the discharge piping from the pump discharge of these systems to the last check valve shall be filled.

4.5 SURVEILLANCE REQUIREMENTS

H. Maintenance of Filled Discharge Pipe

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

3.5 LIMITING CONDITION FOR OPERATION

Average Planar LHGR

During steady state power operation, the Average Planar Linear Heat Generation Rate (APLHGR) of all the rods in any fuel assembly, as a function of average planar exposure, at any axial location, shall not exceed the maximum average planar LHGR shown in Figure 3.5.1. If at any time during operation it is determined by normal surveillance that the limiting value for APLHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the APLHGR is not returned 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.

Amendment No. 58

4.5 SURVEILLANCE REQUIREMENT

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 $\geq$ 25% rated thermal power.

3.5 LIMITING CONDITION FOR OPERATION

I. 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:

$$LHGR_{max} \leq LHGR_{d} \left[1-\left(\frac{\Delta P}{P}\right)_{max}\left(\frac{L}{LT}\right)\right]$$

$LHGR_d$ = Design LHGR = 17.5 Kw/ft. 7x7 fuel

= 13.4 kw/ft for all 8x8 fuel types

$\left(\Delta P/P\right)_{max}$ = Maximum power spiking penalty = 0.037 for 7x7 fuel and 0.0 for 8x8 fuel

LT = Total core length = 12 ft.

L = Axial position above 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.

4.5 SURVEILLANCE REQUIREMENT

J. Linear Heat Generation Rate (LHGR)

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

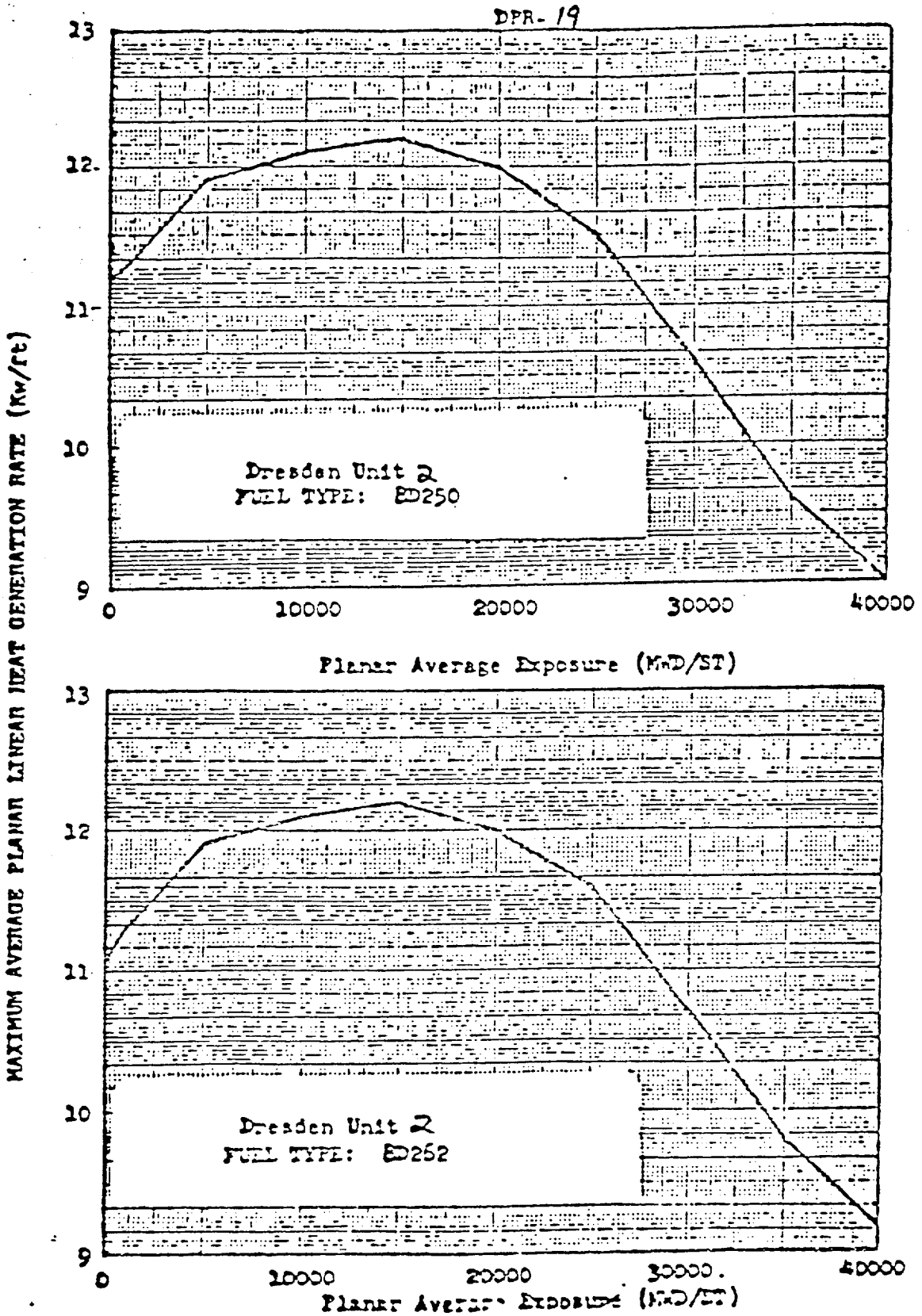
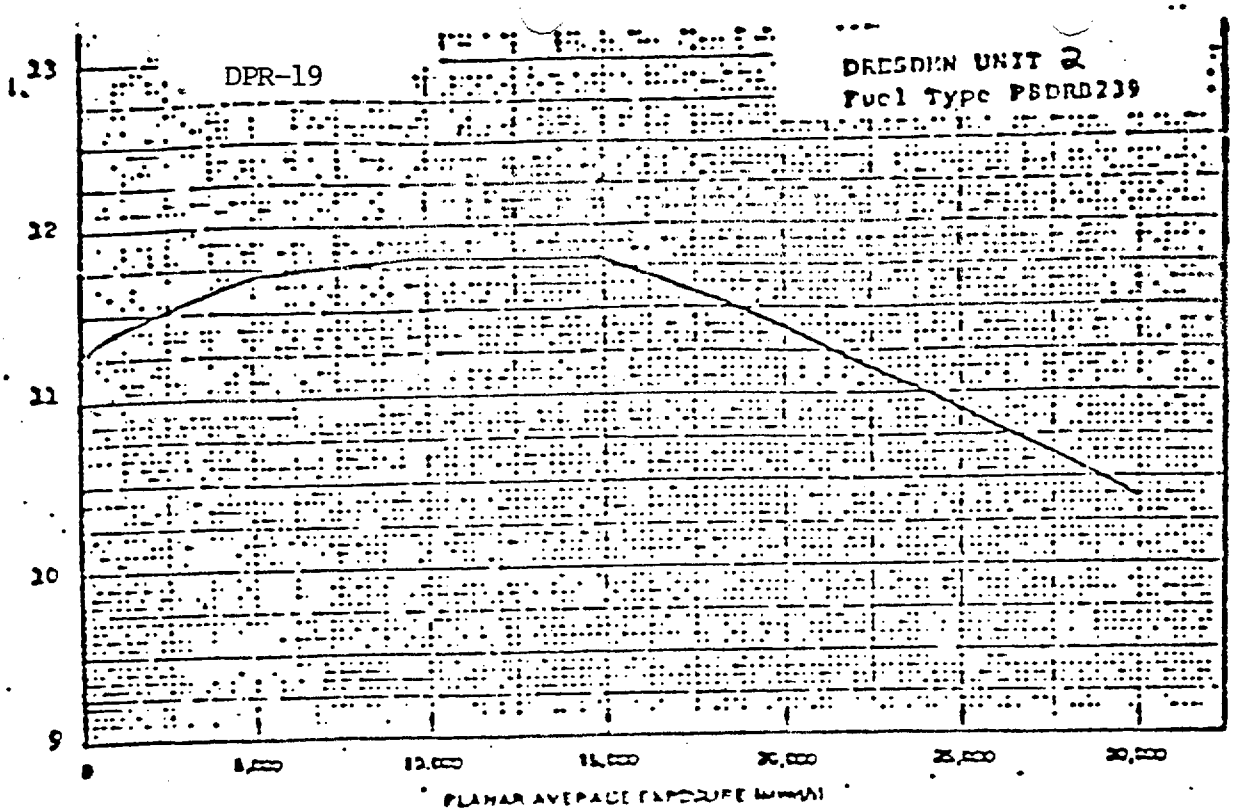
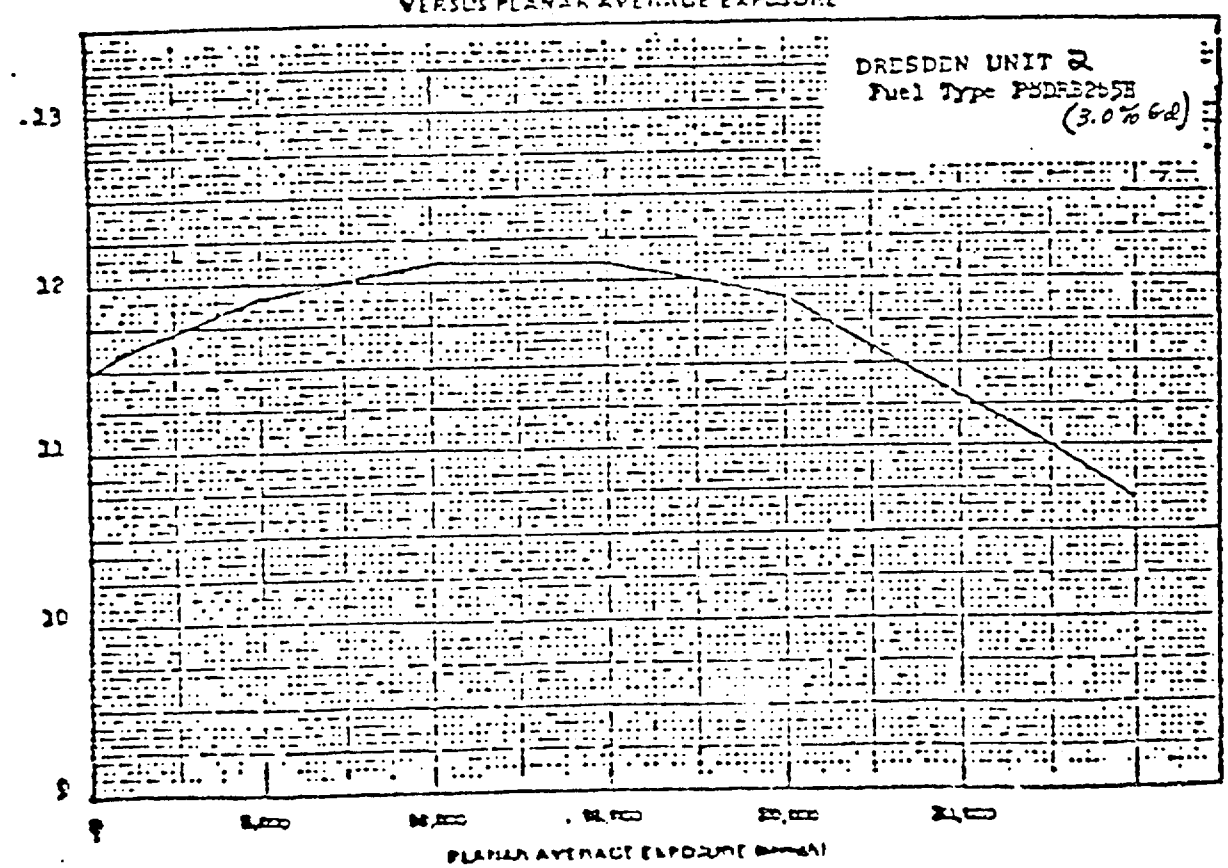


Figure 3.5-1 (Sheet 1 of 5) MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (GRAPHITE) vs. PLANAR AVERAGE EXPOSURE



MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR)
VERSUS PLANAR AVERAGE EXPOSURE



MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR)
VERSUS PLANAR AVERAGE EXPOSURE

Figure 3.5-1
(Sheet 2 of 5)

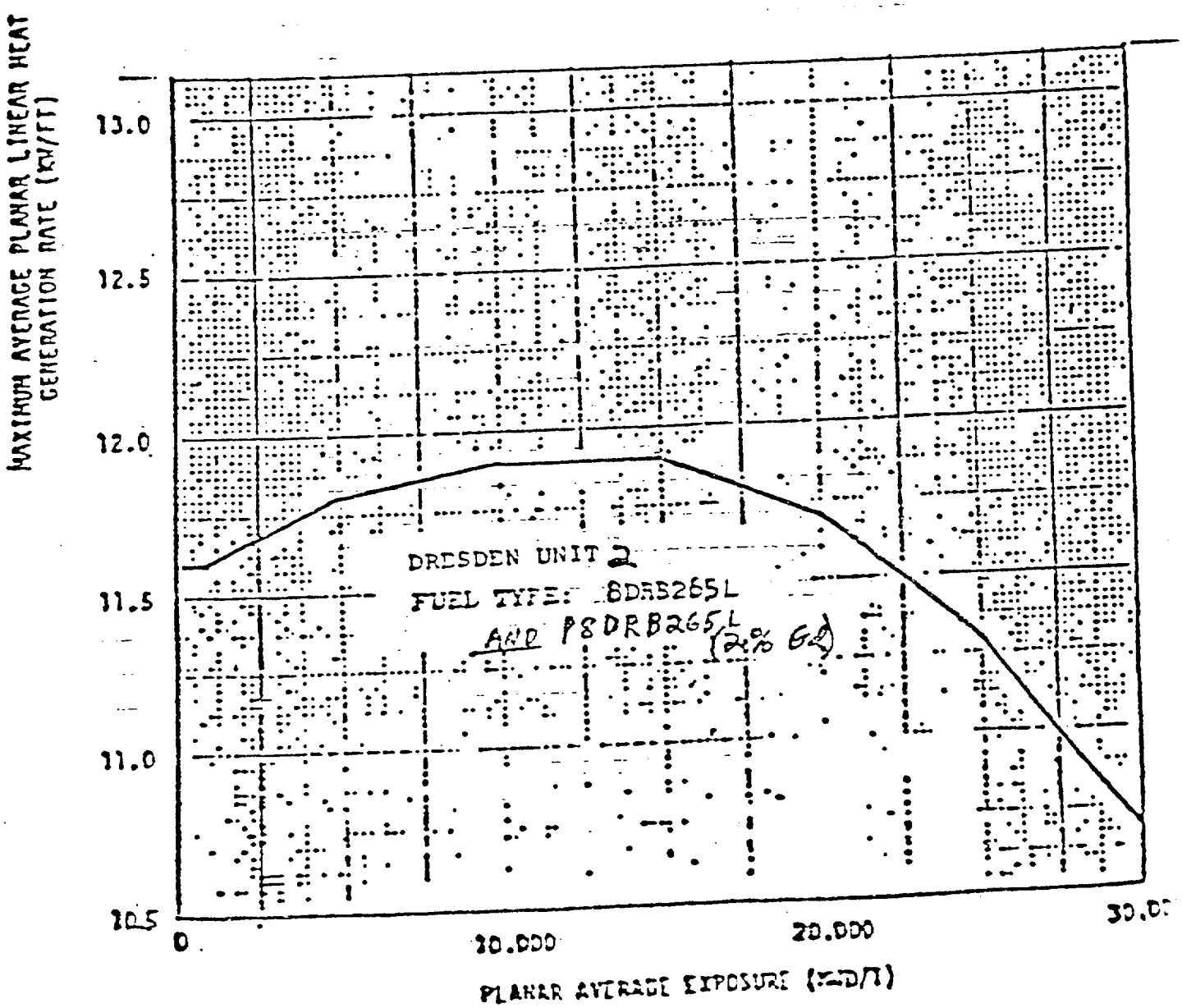
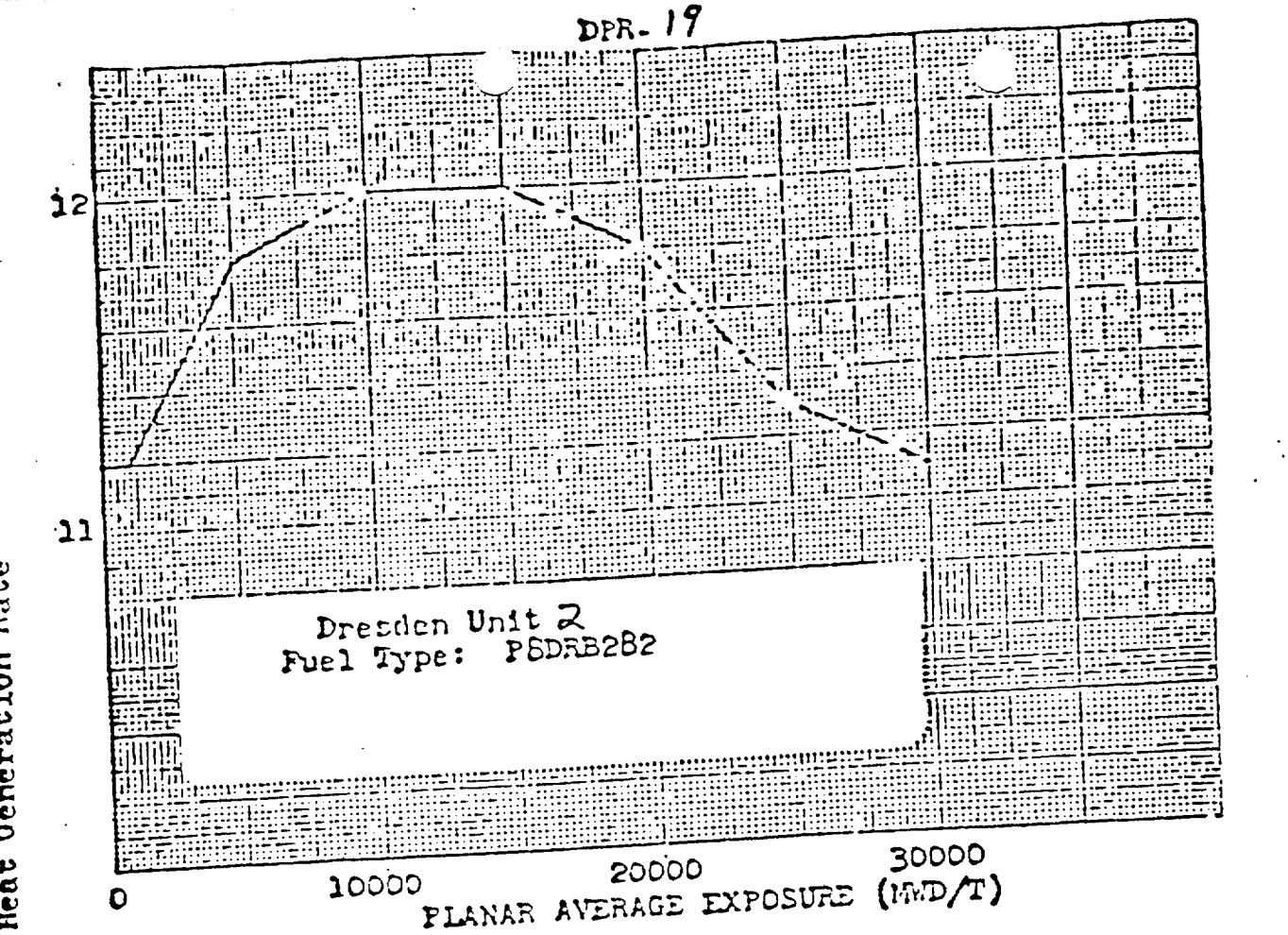
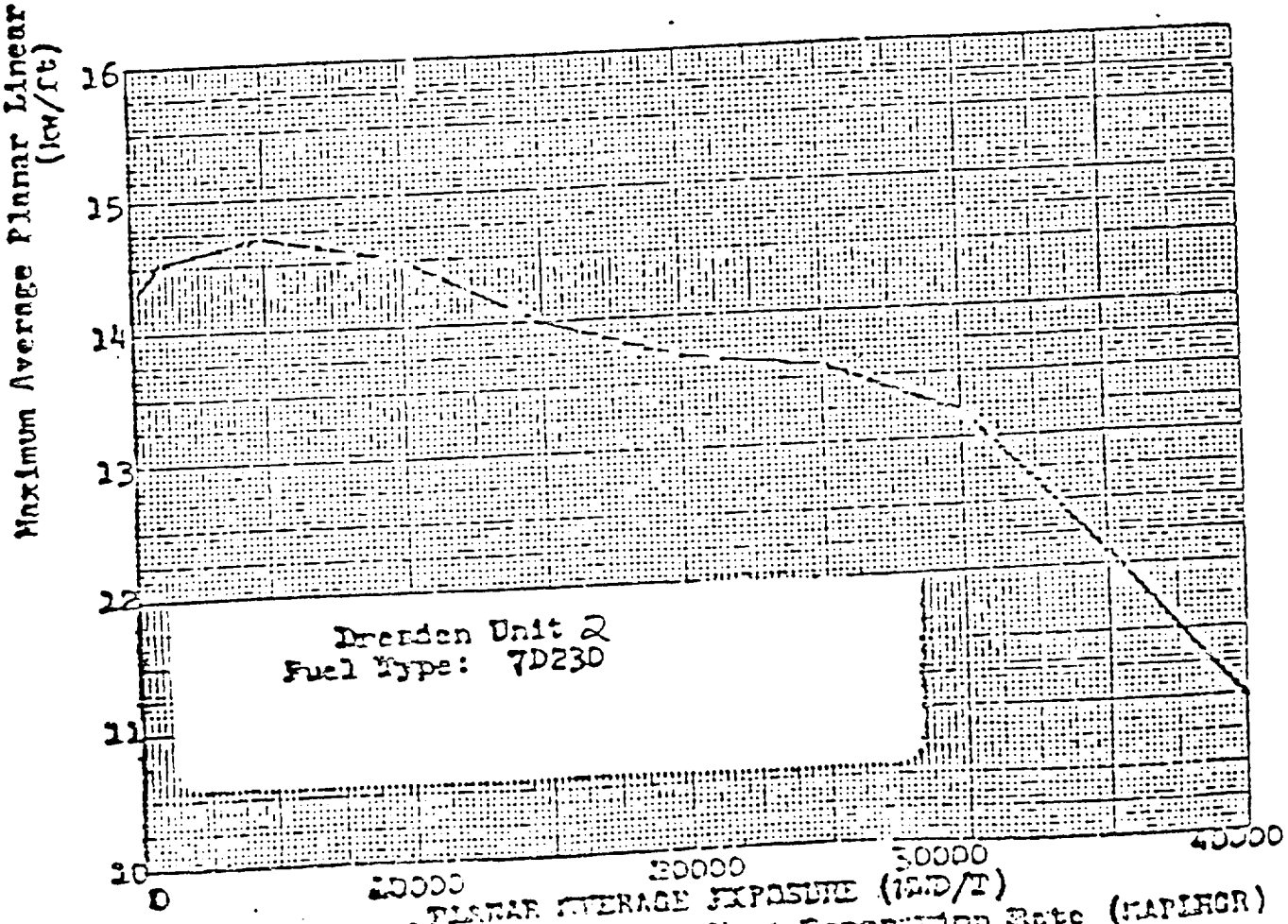


FIGURE 3.5-1 (Sheet 3 of 5) MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VS. PLANAR AVERAGE EXPOSURE





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

Figure 3.5-1 (Sheet 4 of 5)

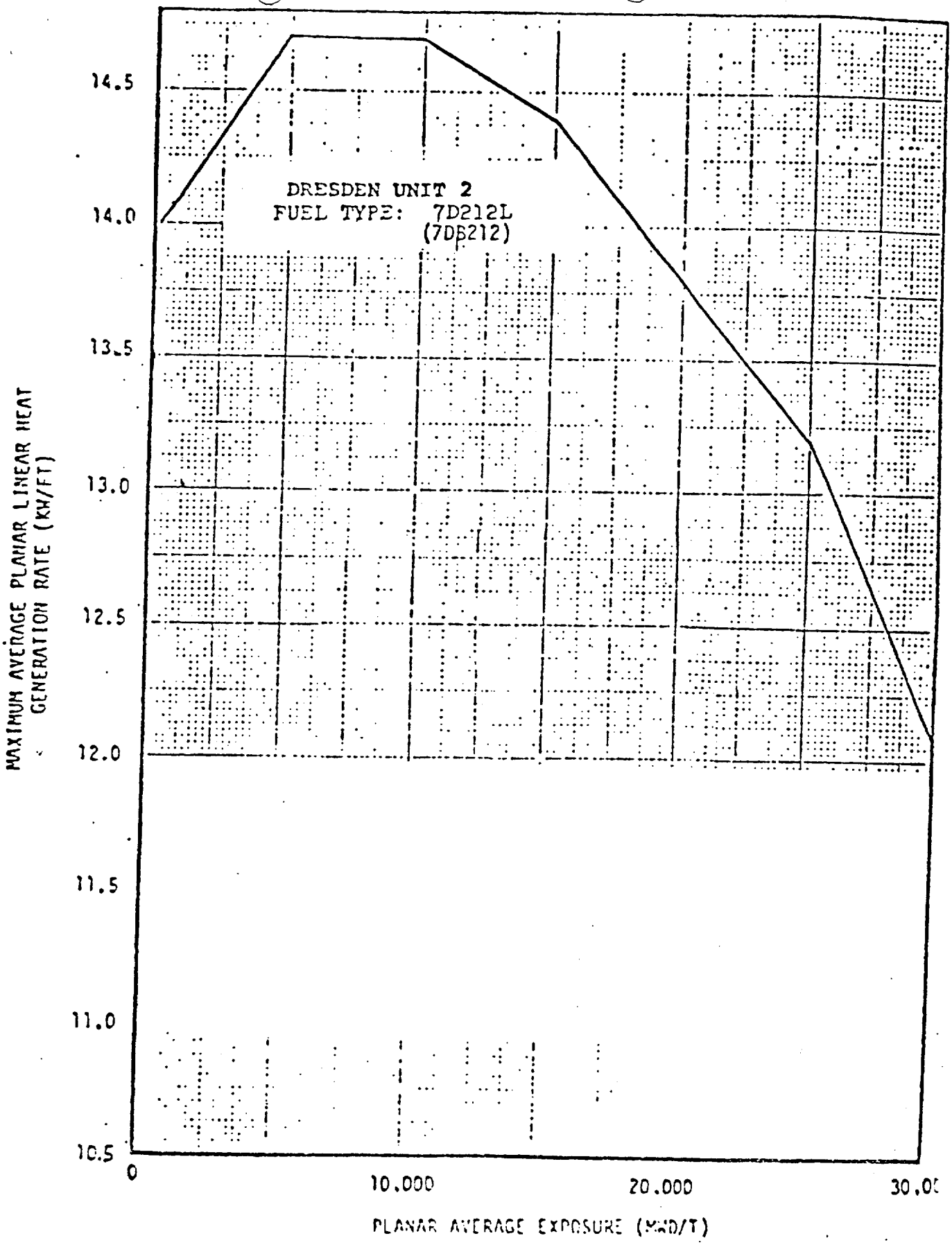


FIGURE 3.5.1 MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VS. PLANAR AVERAGE EXPOSURE (SHEET 5 OF 5)

3.5 LIMITING CONDITION FOR OPERATION

K. Minimum Critical Power Ratio (MCPR)

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

Unit 2

1.24 (7 x 7 fuel)

1.31 (8 x 8/8x8R/P8x8R fuel)

at rated power and flow. For core flows other than rated, these nominal values of MCPR shall be increased by a factor of Kf, where Kf 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.

4.5 SURVEILLANCE REQUIREMENTS

K. Minimum Critical Power Ratio (MCPR)

MCPR shall be determined daily during a reactor power operation at ≥ 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.

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) "Loss of Coolant Accident Analyses Report for Dresden Units 2, 3 and Quad-Cities Units 1, 2 Nuclear Power Stations," NEDO-24146A, Revision1, April 1979.

(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.

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 (1). 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.

(1) "Loss of Coolant Accident Analyses Report for Dresden Units 2, 3 and Quad-Cities Units 1, 2 Nuclear Power Stations," NEDO-24146A, Revision 1, April, 1979.

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

.5 Limiting Condition for Operation Bases (Cont'd)

heat generation rate even if fuel pellet densification is postulated. The power spike penalty is discussed in Reference 2 and assumes a linearly increasing variation in axial gaps between core bottom and top, and assumes with 95% confidence, that no more than one fuel rod exceeds the design LHGR due to power spiking.

K. Minimum Critical Power Ratio (MCPR)

The steady state values for MCPR specified in this Specification were selected to provide margin to accommodate transients and uncertainties in monitoring the core operating state as well as uncertainties in the critical power correlation itself. These values also assure that operation will be such that the initial condition assumed for the LOCA analysis, plus 2% uncertainty 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 value of MCPR stated in this specification for the limiting condition of operation bounds the initial value of MCPR assumed to exist prior to the initiation of the transients. This initial condition which is used in the transient analyses, will preclude violation of the MCPR fuel cladding integrity safety limit. Assumptions and methods used in calculating the required steady state MCPR limit for each reload cycle are documented in Reference 2. The results apply with increasing conservatism while operating with MCPRs greater than specified.

The most limiting transients with respect to MCPR are generally:

a) Rod withdrawal error

b) Load rejection or turbine trip without bypass

c) Loss of feedwater heater

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

For core flow rates less than rated, the steady state MCPR is increased by the formula given in the Specification. This 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) "Generic Reload Fuel Application," NEDE-24011-P-A*

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

4.5 Surveillence Requirements Bases (cont'd)

I. Average Planar LHGR

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

J. Local LHGR

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

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 requirement by a considerable margin. With this low void content, any inadvertent core flow increase would only place operation in a more conservative mode relative to MCPR.

The daily requirement for calculating MCPR above 25 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.

3.6 LIMITING CONDITION FOR OPERATION

an orderly shutdown shall be initiated and the reactor shall be in a Cold Shutdown condition within 24 hours.

E. Safety and Relief Valves

1. During reactor power operating conditions and whenever the reactor coolant pressure is greater than 90 psig and temperature greater than 320°F, all nine of the safety valves shall be operable. The solenoid activated pressure valves shall be operable as required by Specification 3.5.D.

2. If Specification 3.6.E.1 is not met, an orderly shutdown shall be initiated and the reactor coolant pressure and temperature shall be less than or equal to 90 psig and less than or equal to 320°F within 24 hours.

4.6 SURVEILLANCE REQUIREMENT

E. Safety and Relief Valves

A minimum of ⅓ of all safety valves shall be bench checked or replaced with a bench checked valve each refueling outage. The popping point of the safety valves shall be set as follows:

Number of Valves	Set Point (psig)
1	1115*
2	1240
2	1250
2	1260
2	1260

The allowable set point error for each valve is ±1%.

All relief valves shall be checked for set pressure each refueling outage. The set pressures shall be:

Number of Valves	Set Point (psig)
1	1115*
2	Less than or equal to 1130
2	Less than or equal to 1135

*Target Rock combination safety/relief valve

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 58

TO PROVISIONAL OPERATING LICENSE NO. DPR-19

COMMONWEALTH EDISON COMPANY

DRESDEN NUCLEAR POWER STATION, UNIT NO. 2

DOCKET NO. 50-237

1.0 Introduction

By letter dated January 28, 1981 (Reference 1), as supplemented by Reference 8, Commonwealth Edison (CE), the licensee, proposed amendments to Dresden Unit 2 License and Appendix A Technical Specifications. CE has proposed these amendments to support its review of future reloads for Dresden Unit 2 under the provisions of 10 CFR 50.59.

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

2.0 Evaluation

Safety Limit Critical Power Ratio (SLMCPR)

This change provides SLMCPRs in the Technical Specifications for all currently approved core loadings. With retrofit 8x8 fuel in the core the SLMCPR limit is specified as 1.07. Without retrofit 8x8 fuel the SLMCPR limit is 1.06. These limits have previously been found acceptable for this use in Reference 4 and on this basis the proposed change is acceptable.

Rod Drop Accident (RDA) Design Limit

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

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Maximum Average Planar Linear Heat Generation Rate (MAPLHGR)

New MAPLHGR curves reflecting the improved flooding characteristics of retrofit 8x8 fuel have been proposed by the licensee. Curves for 8x8, 8x8 retrofit, and 7x7 fuel of the various enrichments anticipated for future Dresden 2 reloads and extending to burnups of 40,000 MWd/t have been proposed (References 1 and 6).

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

Based on our previous approval of MAPLHGR curves reflecting 8x8 retrofit fuel reflood characteristics (Reference 10) and extension of burnup to 40,000 MWd/t (Reference 9), the licensee's proposed changes are acceptable.

Power Peaking

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

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

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

The LTPF can be expressed as the design linear heat generation rate divided by the plant rated thermal power per unit length of fuel rod. In a similar manner the TPF can be expressed as the maximum linear heat generation rate divided by the plant operating power per unit length of fuel rod. From these definitions it is easily determined that the ratio LTPF/TPF is the ratio of the design linear heat generation rate to the maximum linear heat generation rate times the fraction of rated thermal power, or 1/MFLPD*FRP. Thus, FRP/MFLPD and LTPF/TPF are equivalent. This change in terminology has been approved for other BWRs (References 2 and 3).

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

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

Overpressure Protection Margin to Safety Valve Setpoint

CE has proposed to delete the portion of the license restriction that requires reactor power level restrictions to maintain pressure margin to safety valve (SV) setpoints during the worst case pressurization transient. This restriction was imposed by the licensee to avoid an extensive outage in the event of SV discharge to the drywell. Our criteria for overpressurization protection (Standard Review Plan 5.2.2, NUREG-79/087) have been that "for the design basis normal operational transients, relief valve capacity must be sufficient to limit the pressure so as to prevent SV discharge directly to the containment," and "for the most severe abnormal operational transient, with reactor scram, the SV capacity should be sufficient to limit the pressure to less than 110% of the reactor coolant pressure boundary design pressure." These criteria are satified by the proposed change.

Further, we do not consider the SV discharge to the drywell a safety concern, since all safety systems are to be qualified for LOCA environment which is more severe than the possible SV discharge. We have also reviewed BWR pressure relief systems operating experience (NUREG-0462) and have found that operating experience with SVs has been essentially failure free.

Coastdown Feedwater Heater Restrictions

The licensee has proposed that the license restriction include a requirement to perform a safety evaluation if off-normal feedwater heater operation is needed. We consider this restriction appropriate.

Reactor Protection System (RPS) Delay Time

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

definition of a 50 msec delay time in the licensing analysis. The staff has confirmed that the licensee has in place the capability for demonstrating compliance with the more restrictive specification. The proposed change is acceptable.

Typographical Corrections and Clarification of Bases

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

3.0 Environmental Consideration

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

4.0 Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration; (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner; and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or the health and safety of the public.

Date: Amendment No. 58

References

1. Letter from R.F. Janecek (CECo) to the Director of Nuclear Reactor Regulation (USNRC), dated January 28, 1981.

2. Letter from T.A. Ippolito to D.L. Peoples (CECo), dated April 16, 1980 transmitting Amendment No. 42 to Operating License No. DPR-25 for Dresden Nuclear Power Station Unit No. 3.

3. Letter from T.A. Ippolito (USNRC) to G.T. Berry (Power Authority of the State of New York), dated November 22, 1978.

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

5. Letter from D.G. Eisenhut (USNRC) to R. Gridley (GE), dated June 9, 1978.

6. "Loss-of-Coolant Accident Analysis Report for Dresden Units 2, 3 and Quad Cities Units 1 and 2 Nuclear Power Stations", NEDO-24164 A, dated April 1979.

7. Memorandum from R.L. Baer to D.L. Ziemann, "Evaluation of Quad Cities Unit 2 for Cycle 4 Operation", dated February 21, 1978.

8. Letter from J.S. Abel (CECo) to the Director of Nuclear Reactor Regulation (USNRC), dated February 23, 1981.

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

10. Letter from D.L. Ziemann (USNRC) to Cordell Reid (CECo), dated April 24, 1979.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-237

COMMONWEALTH EDISON COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO PROVISIONAL OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 58 to Provisional Operating License No. DPR-19 issued to Commonwealth Edison Company (the licensee), which revised the license and Technical Specifications for operation of the Dresden Nuclear Power Station, Unit No. 2, located in Grundy County, Illinois. The amendment is effective as of the date of issuance.

The amendment (1) authorizes changes to the Technical Specifications to support review of future reloads for Dresden Unit 2 under provisions of 50.59 and (2) modifies license condition 3.F to assure a conservative MCPR operating limit during coastdown operation.

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

The Commission has determined that the issuance of the amendment will not result in any significant environmental impact and that pursuant to 10 CFR Section 51.5(d)(4) an environmental impact statement, or negative

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

For further details with respect to this action, see (1) the application for amendment dated January 28, 1981, (2) Amendment No. 58 to License No. DPR-19, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, NW, Washington, D.C., and at the Morris Public Library, 604 Liberty Street, Morris, Illinois. A copy of items (2) and (3) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 31st day of March, 1981.

FOR THE NUCLEAR REGULATORY COMMISSION

Dennis M. Crutchfield

Dennis M. Crutchfield, Chief
Operating Reactors Branch #5
Division of Licensing