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

January 17, 1985

Docket No. 50-237
LS05-85-01-017

Mr. Dennis L. Farrar
Director of Nuclear Licensing
Commonwealth Edison Company
Post Office Box 767
Chicago, Illinois 60690

Dear Mr. Farrar:

SUBJECT: TECHNICAL SPECIFICATION CHANGES RELATING TO THE CYCLE 10 RELOAD

Re: Dresden Nuclear Power Station, Unit No. 2

The Commission has issued the enclosed Amendment No. 84 to Provisional Operating License No. DPR-19 for the Dresden Nuclear Power Station, Unit No. 2. The amendment consists of changes to the Technical Specifications in response to your applications dated September 11, 27 and 28, 1984 and October 2, 1984.

The amendment authorizes changes to the Technical Specifications to support Cycle 10 operation of Dresden 2 with reload fuel supplied by and the associated analyses performed by Exxon Nuclear Company. The amendment also authorizes Dresden 2 to use hafnium in control rod blades and, specifically, to install General Electric hybrid design hafnium control rod assemblies starting in Cycle 10, provides new limiting conditions for operation and surveillance requirements for a newly modified scram system having improved reliability and changes the calibration and functional test frequencies for certain specific instruments that are being modified into analog trip systems. Specifically related to the operation with the reload fuel, the amendment authorizes extension of the MAPLHGR curves for 8 x 8 and 9 x 9 (LTA) fuel types and for GE P8DRB265H fuel type and deletes the MAPLHGR curve for GE fuel type P8DRB239 which has never been used at Dresden and is not expected to be in the future.

Notices of Consideration of Issuance of Amendment to License and Proposed No Significant Hazards Consideration Determination and Opportunity for Hearing related to the requested actions were published in the Federal Register on October 24, 1984 (49 FR 42815) and November 21, 1984 (49 FR 45944 and 45945). No requests for hearing and no comments were received.

A copy of our related Safety Evaluation is also enclosed. This action will appear in the Commission's Monthly Notice publication in the Federal Register.

Sincerely,

Original signed by
John A. Zwolinski, Chief
Operating Reactors Branch #5
Division of Licensing

Enclosures:
1. Amendment No. 84 to DPR-19
2. Safety Evaluation

cc w/enclosures:
See next page

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

COMMONWEALTH EDISON COMPANY

DRESDEN NUCLEAR POWER STATION, UNIT NO. 2

DOCKET NO. 50-237

AMENDMENT TO PROVISIONAL OPERATING LICENSE

Amendment No. 84
License No. DPR-19

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

   A. The applications for amendment by the Commonwealth Edison Company (the licensee) dated September 11, 27 and 28, 1984 and October 2, 1984 comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   B. <u>Technical Specifications</u>

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 84, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

John A. Zwolinski, Chief
Operating Reactors Branch #5
Division of Licensing

Attachment:
Changes to the Technical
  Specifications

Date of Issuance: January 17, 1985

ATTACHMENT TO LICENSE AMENDMENT NO. 84

PROVISIONAL OPERATING LICENSE NO. DPR-19

DOCKET NO. 50-237

Revise the Technical Specifications by replacing the following pages, which reflect the pagination of Amendment 83, with the attached pages. The revised pages contain the captioned amendment number and marginal lines to reflect the area of change. Some of the changes are on Radiological Effluent Technical Specifications pages which are not effective until March 15, 1985. These pages have a previous captioned amendment number of 83. The changes marked by marginal lines on these pages are effective as of the date of this amendment.

REMOVE	INSERT
iii (1)	iii (2)
v (1)	v (2)
vii (1)	vii (2)
viii (1)	viii (2)
3/4.1-5	3/4.1-5
3/4.1-8 through 3/4.1-10	3/4.1-8 through 3/4.1-10
B 3/4.1-12	B 3/4.1-12
B 3/4.1-19 and B 3/4.1-20	B 3/4.1-19 and B 3/4.1-20
3/4.2-12 and 3/4.2-13 (1)	3/4.2-12 and 3/4.2-13 (2)
3/4.2-17 (1)	3/4.2-17 (2)
3/4.2-19 (1)	3/4.2-19 (2)
B 3/4.2-32 and B 3/4.2-33 (1)	B 3/4.2-32 and B 3/4.2-34 (2)
3/4.5-15	3/4.5-15
3/4.5-17 through 3/4.5-27	3/4.5-17 through 3.4.5-30
B 3/4.5-28 through B 3/4.5-41	B 3/4.5-31 through B 3/4.5-44
5.1	5.1

(1) Pages which are from Amendment 83.

(2) Pages which contain information from Amendment 83 but also have changes approved in this amendment which are immediately effective.

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List of Tables

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TABLE 3.1.1

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENTS

Minimum Number Operable Inst. Channels per Trip (1) System	Trip Function	Trip Level Setting	Modes in Which Function Must be Operable: Refuel (7)	Startup/Hot Standby	Run	Action*
1	Mode Switch in Shutdown		X	X	X	A
1	Manual Scram		X	X	X	A
	IRM					
3	High Flux	(LT/E) 120/125 of Full Scale	X	X	X(5)	A
3	Inoperative		X	X	X(5)	A
	APRM					
2	High Flux	Specification 2.1.A.1	X	X(9)	X	A or B
2	Inoperative**		X	X(9)	X	A or B
2	Downscale	(GT/E) 5/125 of Full Scale	X(12)	X(12)	X(13)	A or B
2	High Flux (15% Scram)	Specification 2.1.A.2	X	X	X(14)	A
2	High Reactor Pressure	(LT/E) 1060 psig	X(11)	X	X	A
2	High Drywell Pressure	(LT/E) 2 psig	X(8), X(10)	X(8), (10)	X(10)	A
2	Reactor Low Water Level	(GT/E) 1 inch***	X	X	X	A
2 (Per Bank)	High Water Level in Scram Discharge Volume (Thermal and dP Switch)	(LT/E) 40 inches above bottom of the Instrument Volume	X(2)	X	X	A or D
2	Turbine Condenser Low Vacuum	(GT/E) 23 in. Hg Vacuum	X(3)	X(3)	X	A or C
2	Main Steam Line High Radiation	(LT/E) 3 X Full Power Background	X(3)	X(3)	X(15)	A or C
4(6)	Main Steam Line Isolation Valve Closure	(LT/E) 10% Valve Closure	X(3)	X(3)	X	A or C
2	Generator Load Rejection	****	X(4)	X(4)	X(4)	A or C
2	Turbine Stop Valve Closure	(LT/E) 10% Valve Closure	X(4)	X(4)	X(4)	A or C
2	Turbine Control - Loss of Control Oil Pressure	(GT/E) 900 psig	X	X	X	A or C

Notes: (LT/E) = Less than or equal to.
(GT/E) = Greater than or equal to.
(Notes continue on next two pages)

TABLE 4.1.1

SCRAM INSTRUMENTATION FUNCTIONAL TESTS

MINIMUM FUNCTIONAL TEST FREQUENCIES FOR SAFETY INSTR. AND CONTROL CIRCUITS

Instrument Channel	Group (3)	Functional Test	Minimum Frequency (4)
Mode Switch in Shutdown	A	Place Mode Switch in Shutdown	Each Refueling Outage
Manual Scram	A	Trip Channel and Alarm	Every 3 Months
IRM			
* High Flux	C	Trip Channel and Alarm (5)	Before Each Startup (6)
* Inoperative	C	Trip Channel and Alarm	Before Each Startup (6)
APRM			
High Flux	B	Trip Output Relays (5)	Once Each Week
Inoperative	B	Trip Output Relays	Once Each Week
Downscale	B	Trip Output Relays (5)	Once Each Week
High Flux (15% scram)	B	Trip Output Relays	Before Each Startup
High Reactor Pressure	A	Trip Channel and Alarm	(1)
High Drywell Pressure	A	Trip Channel and Alarm	(1)
Reactor Low Water Level (2)	B	(8)	(1)
High Water Level in Scram Discharge Volumes (Thermal and dp Switch)	A	Trip Channel and Alarm (7)	Every 3 Months
Turbine Condenser Low Vacuum	A	Trip Channel and Alarm	(1)
Main Steam Line High Radiation (2)	B	Trip Channel and Alarm (5)	Once Each Week
Main Steam Line Isolation Valve Closure	A	Trip Channel and Alarm	(1)
Generator Load Rejection	A	Trip Channel and Alarm	(1)
Turbine Stop Valve Closure	A	Trip Channel and Alarm	(1)
Turbine Control - Loss of Control Oil Pressure	A	Trip Channel and Alarm	(1)

Notes: (See next page.)

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NOTES: (For Table 4.1.1)

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 steam line 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 (1.0.G). This Instrument Function 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.

7. Only the electronics portion of the thermal switches will be tested using an electronic calibrator during the three month test. A water column or equivalent will be used to test the dp switches.

8. A functional test of the master and slave trip unit is required monthly (staggered one channel out of 4 every week). A calibration of the trip unit is to be performed concurrent with the functional testing.

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TABLE 4.1.2
SCRAM INSTRUMENTATION CALIBRATIONS
MINIMUM CALIBRATION FREQUENCIES FOR REACTOR PROTECTION INSTRUMENT CHANNELS

Instrument Channel	Group (1)	Calibration Test	Minimum Frequency (2)
* High Flux IRM	C	Comparison to APRM after Heat Balance	Every Shutdown (4)
High Flux APRM			
Output Signal	B	Heat Balance	Once Every 7 Days
Flow Bias	B	Standard Pressure and Voltage Source	Refueling Outage
High Reactor Pressure	A	Standard Pressure Source	Every 3 Months
High Drywell Pressure	A	Standard Pressure Source	Every 3 Months
Reactor Low Water Level	B	Water Level	(5)
Turbine Condenser Low Vacuum	A	Standard Vacuum Source	Every 3 Months
Main Steam Line High Radiation	B	Standard Current Source (3)	Every 3 Months
Turbine Control - Loss of Control Oil Pressure	A	Pressure Source	Every 3 Months
High Water Level in Scram Discharge Volume (dp only)	A	Water Level	Once per Refueling Outage

NOTES: (For Table 4.1.2)

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

2. Calibration 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.

3. The current source provides an instrument channel alignment. Calibration using a radiation source shall be made during each refueling outage.

*4. If reactor startups 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.

5. Trip units are calibrated monthly concurrently with functional testing (staggered one channel out of 4 every week). Transmitters are calibrated once per operating cycle.

3.1 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

system can tolerate a single failure and still perform its intended function of scramming the reactor. Three APRM instrument channels are provided for each protection trip system.

APRM's #1 and #3 operate contacts in a one subchannel and APRM's #2 and #3 operate contacts in the other subchannel. APRM's #4, #5 and #6 are arranged similarly in the other protection trip system. Each protection trip system has one more APRM than is necessary to meet the minimum number required per channel. This allows the bypassing of one APRM per protection trip system for maintenance, testing or calibration. Additional IRM channels have also been provided to allow for bypassing of one such channel. The bases for the scram settings for the IRM, APRM, high reactor pressure, reactor low water level, generator load rejection, and turbine stop valve closure are discussed in Specification 2.3.

Instrumentation (pressure switches) in the drywell are provided to detect a loss of coolant accident and initiate the emergency core cooling equipment. This instrumentation is a backup to the water level instrumentation which is discussed in Specification 2.2. A scram is provided at the same setting as the emergency core cooling system (ECCS) initiation to minimize the energy which must be accommodated during a loss of coolant accident and to prevent the reactor from going critical following the accident.

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 system is an individual instrument volume for each of the south and north CRD accumulators. These two volumes and their piping can hold in excess of 90 gallons of water and is the low point in the piping. No credit was taken for these volumes in the design of the discharge piping relative to the amount of water which must be accommodated during a scram. During normal operations, the discharge volumes are empty; however, should either volume fill with water, the water discharged to the piping from the Reactor may not be accommodated which could result in slow scram times or partial or no control rod insertion. To preclude this occurrence, level switches have been installed in both volumes which will alarm and scram the Reactor when the volume remaining in either instrument volume is approximately 40 gallons. For diversity of level sensing methods that will ensure and provide a scram, both differential pressure switches and thermal switches have been incorporated into the design and logic of the system. The setpoint for the scram signal has been chosen on the basis of providing sufficient volume remaining to accommodate a scram even with 5 gpm leakage per drive into the SDV. As indicated above, there is sufficient volume in the piping to accommodate the scram without impairment of the scram times or the 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 properly.

4.1 SURVEILLANCE REQUIREMENT BASES (Cont'd.)

Reactor low water level instruments 2-263-57A, 2-263-57B, 2-263-58A, and 2-263-58B have been modified to be an analog trip system. The analog trip system consists of an analog sensor (transmitter) and a master/slave trip unit setup which ultimately drives a trip relay. The frequency of calibration and functional testing for instrument loops of the analog trip system, including reactor low water level, has been established in Licensing Topcial Report NEDO-21617-A (December 1978). With the one-out-of-two-taken-twice logic, NEDO-21617-A states that each trip unit be subjected to a calibration/functional test of one month (staggered one channel out of four every week). An adequate calibration/ surveillance test interval for the transmitter is once per operating cycle.

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 0.4% would occur and thus provide 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.

4.1 SURVEILLANCE REQUIREMENT BASES (Cont'd.)

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 Volume dp and Thermal Switches, 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 for fuel fabricated by GE shall be checked once per day to determine if the APRM gains or scram requires adjustment. This may normally be done by checking the LPRM readings, TIP traces, or process computer calculations.

Only a small number of control rods are moved daily and thus the peaking factors are not expected to change significantly and thus a daily check of the MFLPD is adequate.

For fuel fabricated by ENC, the power distribution will be checked once per day to ensure consistency with the power distribution assumptions of the fuel design analysis for overpower conditions. During periods of operation beyond these power distribution assumptions, the APRM gains or scram settings may be adjusted to ensure consistency with the fuel design criteria for overpower conditions.

TABLE 3.2.3

INSTRUMENTATION THAT INITIATES ROD BLOCK

Minimum No. of Operable Inst. Channels Per Trip System (1)	Instrument	Trip Level Setting
1	APRM upscale (flow bias) (7)	Less than or equal to (0.58 W_D plus 50) (FRP/MFLPD) (See Note 2)
1	APRM upscale (refuel and Startup/Hot Standby mode)	Less than or equal to 12/125 full scale
2	APRM downscale (7)	Greater than or equal to 3/125 full scale
1	Rod block monitor upscale (flow bias) (7)	Less than or equal to (0.65 W_D plus 45) (see Note 2)
1	Rod block monitor downscale (7)	Greater than or equal to 5/125 full scale
3	IRM downscale (3)	Greater than or equal to 5/125 full scale
3	IRM upscale	Less than or equal to 108/125 full scale
3	IRM detector not fully inserted in the core	N/A
2 (5)	SRM detector not in startup position	(4)
2 (5) (6)	SRM upscale	Less than or equal to 10^5 counts/sec.
1 (per bank)	Scram discharge volume water level - high	(LT/E) 26 inches above the bottom of the instrument volume

Notes:
(See Next Page)

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TABLE 3.2.3 (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. A 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. For the scram discharge volume water level high rod block, there is one instrument channel per bank.

2. W_D percent of drive flow required to produce a rated core flow of 98 Mlb/hr. MFLPD = highest value of FLPD for G.E. fuel.

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

4. This function may be bypassed when the count rate is greater than or equal to 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 test at atmospheric pressure during or after refueling at power levels not to exceed 5 MWt.

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Instrument Channel	Instrument Functional Test	Calibration	Instrument Check
ECCS Instrumentation			
1. Reactor Low-Low Water Level	(1)	Once/3 Months	Once/Day
2. Drywell High Pressure	(1)	Once/3 Months	None
3. Reactor Low Pressure	(1)	Once/3 Months	None
4. Containment Spray Interlock			
a. 2/3 Core Height	(1) (13)	(13)	None
b. Containment High Pressure	(1)	Once/3 Months	None
5. Low Pressure Core Cooling Pump Discharge	(1)	Once/3 Months	None
6. Undervoltage Emergency Bus	Refueling Outage	Refuel Outage	Once/3 months
7. Sustained High Reactor Pressure	(1)	Once/3 Months	None
8. Degraded Voltage Emergency Bus	Refueling Outage (10)	Refuel Outage	Monthly
Rod Blocks			
1. APRM Downscale	(1) (3)	Once/3 Months	None
2. APRM Flow Variable	(1) (3)	Refuel Outage	None
3. APRM Upscale (Startup/Hot Standby)	(2) (3)	(2) (3)	(2)
4. IRM Upscale	(2) (3)	(2) (3)	(2)
5. IRM Downscale	(2) (3)	(2) (3)	(2)
6. IRM Detector Not Fully Inserted in the Core	(2)	N/A	None
7. RBM Upscale	(1) (3)	Refuel Outage	None
8. RBM Downscale	(1) (3)	Once/3 Months	None
9. SRM Upscale	(2) (3)	(2) (3)	(2)
10. SRM Detector Not in Startup Position	(2) (3)	(2) (3)	(2)
11. Scram Instrument Volume Level High	Once/3 Months (9)	None	None
Containment Monitoring			
1. Pressure			
a. Minus 5 in. Hg to plus 5 psig Indicator	None	Once/3 Months	Once/Day
b. 0 to 75 psig Indicator	None	Once/3 Months	None
2. Temperature	None	Refuel Outage	Once/Day
3. Drywell-Torus Differential Pressure (5) (6) (0-3 psid)	None	Once/6 Months (Two Channels Operable) Once/Month (One Channel Operable)	None
4. Torus Water Level (5) (6)	None	Once/6 Months	
a. Plus or minus 25 in. Wide Range Indicator			
b. 18 in. Sight Glass			
Safety/Relief Valve Monitoring			
1. Safety/Relief Valve Position Indicator (Acoustic Monitor) (8)	(7)	None	Once Per 31 Days
2. Safety/Relief Valve Position Indicator (Temperature Monitor (8)	None	Once every 18 months	Once Per 31 Days
3. Safety Valve Position Indicator (Acoustic Monitor) (8)	(7)	None	Once Per 31 Days
4. Safety Valve Position Indicator (Temperature Monitor) (8)	None	Once every 18 months	Once Per 31 Days
Main Steam Line Isolation			
1. Steam Tunnel High Temperature	Refueling Outage	Refuel Outage	None
2. Steam Line High Flow	(1)	Once/3 Months	Once/Day
3. Steam Line Low Pressure	(1)	Once/3 Months	None
4. Steam Line High Radiation	(1) (3)	Once/3 Months (4)	Once/Day
Isolation Condenser Isolation			
1. Steam Line High Flow	(1)	Once/3 Months	None
2. Condensate Line High Flow	(1)	Once/3 Months	None
HPCI Isolation			
1. Steam Line High Flow	(1) (11) (12)	(11) (12)	None
2. Steam Line Area High Temperature	Refueling Outage	Refuel Outage	None
3. Low Reactor Pressure	(1) (13)	(13)	None
Reactor Building Vent Isolation and SBGTS Initiation			
1. Refueling Floor Radiation Monitors	(1)	Once/3 Months	Once/Day

Notes: (See Next Two Pages)

NOTES: (For Table 4.2.1) (Cont'd.)

9. The functional test of the Scram Discharge Volume thermal switches 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 thermal switches in the scram discharge volume tank. Based on the above, no calibration is required for these instrument channels.

10. Functional test shall include verification of the second level undervoltage (degraded voltage) timer bypass and shall verify operation of the degraded voltage 5-minute timer and inherent 7-second timer.

11. Verification of time delay setting between 3 and 9 seconds shall be performed during each refueling outage.

12. Trip units are functionally tested monthly (staggered one channel out of four every week). A calibration of the trip units is to be performed concurrent with the functional testing.

13. Trip units are functionally tested monthly (staggered one division out of two every two weeks). A calibration of the trip units is to be performed concurrent with the functional testing.

4.2 SURVEILLANCE REQUIREMENT BASES (Cont'd.)

A more usual case is that the testing is not done independently. If both channels are bypassed and tested at the same time, the result is shown in Curve No. 3. Note that the minimum occurs at about 40,000 hours, much longer than for cases 1 and 2. Also, the minimum is not nearly as low as Case 2 which indicates that this method of testing does not take full advantage of the redundant channel. Bypassing both channels for simultaneous testing should be avoided.

The most likely case would be to stipulate that one channel be bypassed, tested and restored, and then immediately following, the second channel be bypassed, tested and restored. This is shown by Curve No. 4. Note that there is no true minimum. The curve does have a definite knee and very little reduction in system unavailability is achieved by testing at a shorter interval than computed by the equation for a single channel.

The best test procedure of all those examined is to perfectly stagger the tests. That is, if the test interval is four months, test one or the other channel every two months. This is shown in Curve No. 5. The difference between Cases 4 and 5 is negligible. There may be other arguments, however, that more strongly support the perfectly staggered tests, including reductions in human error.

The conclusions to be drawn are these:

1. A 1 out of n system may be treated the same as a single channel in terms of choosing a test interval; and

2. More than one channel should not be bypassed for testing at any one time.

The analog trip system consists of an analog sensor (transmitter) and a master/slave trip unit setup which ultimately drives a trip relay. The frequency of calibration and functional testing for instrument loops of the analog system, including reactor low water level, has been established in Licensing Topical Report NEDO-21617-A (December, 1978).

For instruments 2(3)-2389A, B, C, D, the one-of-two-taken-twice logic exists, and NEDO-21617-A states that each trip unit be subjected to a calibration/test frequency (staggered one channel out of four per week) of one month. An adequate calibration/surveillance test interval for the transmitter is once per operating cycle.

4.2 SURVEILLANCE REQUIREMENT BASES (Cont'd.)

For instruments 2(3)-263-73A, 73B and 2(3)-2352, 2353, the logic downstream of the output relay contacts exhibits a one-out-of-two logic and, by utilizing the Availability Criteria identified in NEDO-21617-A, each of these trip units should also be subjected to a calibration/test frequency (staggered one division out of two per two weeks) of one month. An adequate calibration/surveillance test interval for the transmitter is once per operating cycle.

The radiation monitors in the ventilation duct and on the refueling floor which initiate building isolation and standby gas treatment operation are arranged in two 1 out of 2 logic systems. The bases given above for the rod blocks applies here also and were used to arrive at the functional testing frequency.

Based on experience at Dresden Unit 1 with instruments of similar design, a testing interval of once every three months has been found to be adequate.

The automatic pressure relief instrumentation can be considered to be a 1 out of 2 logic system and the discussion above applies also.

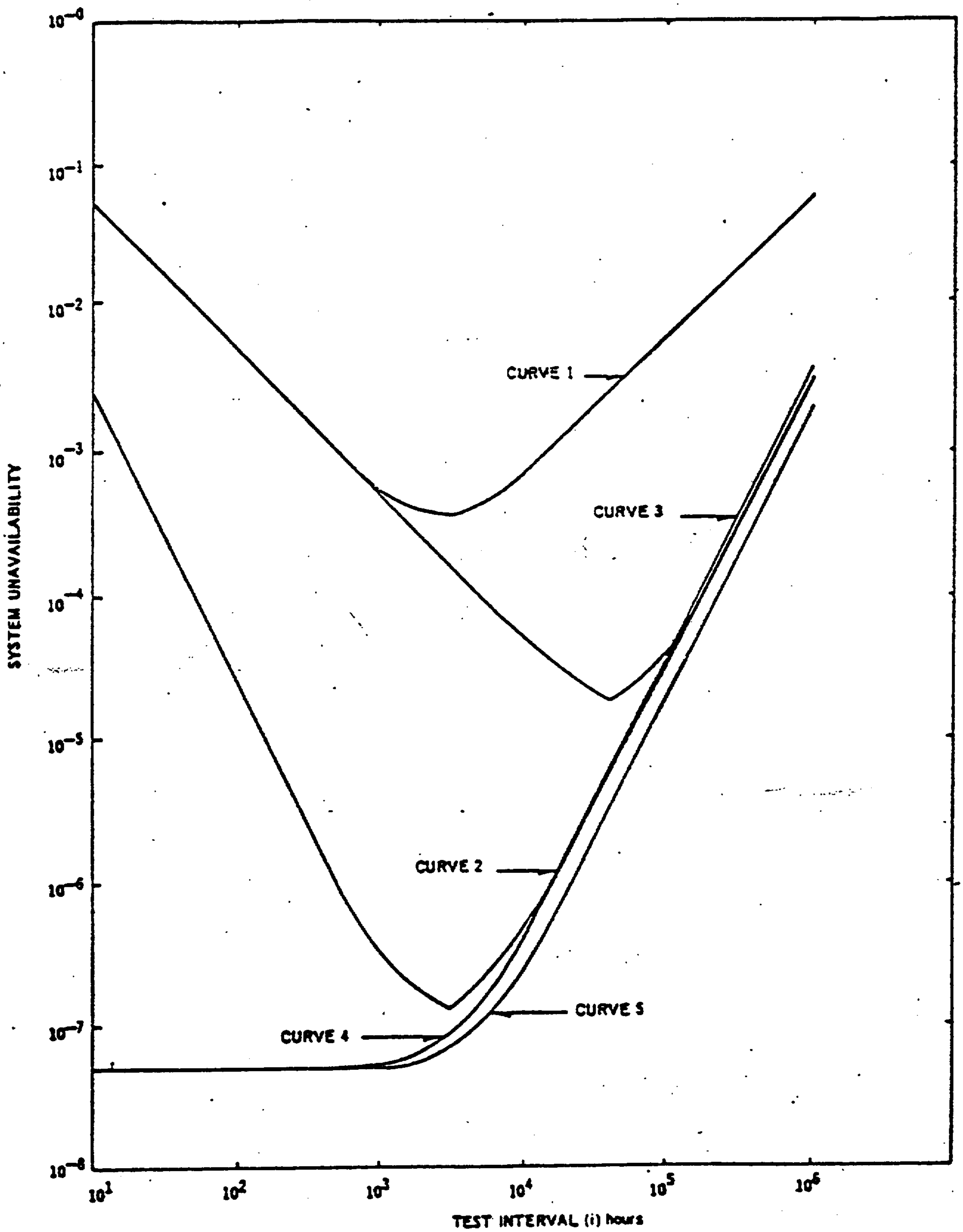


Figure 4.2.2.

TEST INTERVAL VS. SYSTEM UNAVAILABILITY

3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

I. 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 for G.E. fuel and average bundle exposure for Exxon fuel at any axial location, shall not exceed the maximum average planar LHGR shown in Figure 3.5-1 (consisting of eight curves). 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.

J. LOCAL LHGR

During steady state power operation, the linear heat generation rate (LHGR) of any rod in any fuel assembly fabricated

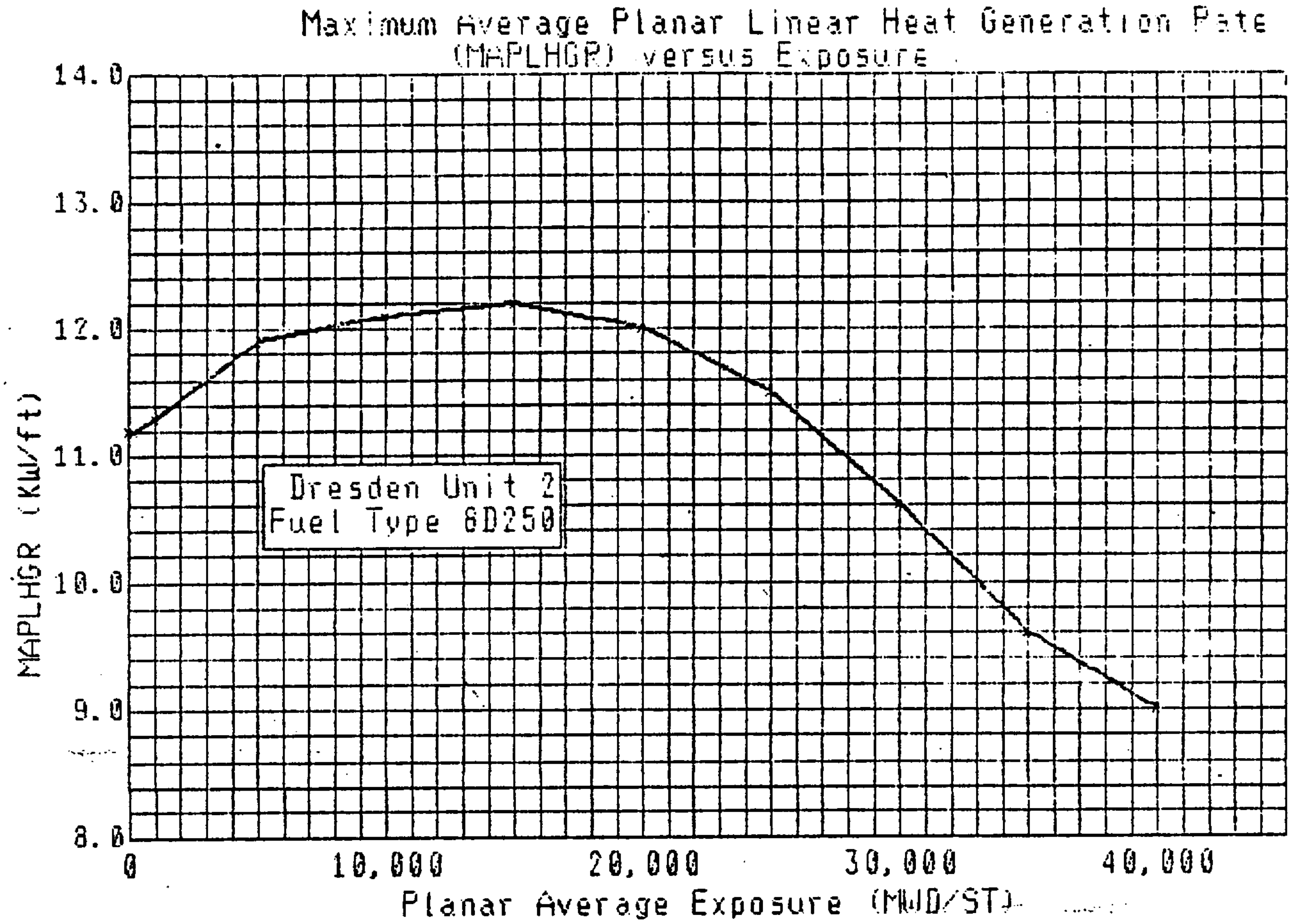
4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

I. Average Planar Linear Heat Generation Rate (APLHGR)

The APLHGR for each type of fuel as a function of average planar exposure for G.E. fuel and average bundle exposure for Exxon fuel shall be determined daily during reactor operation at greater than or equal to 25% rated thermal power.

J. Linear Heat Generation Rate (LHGR)

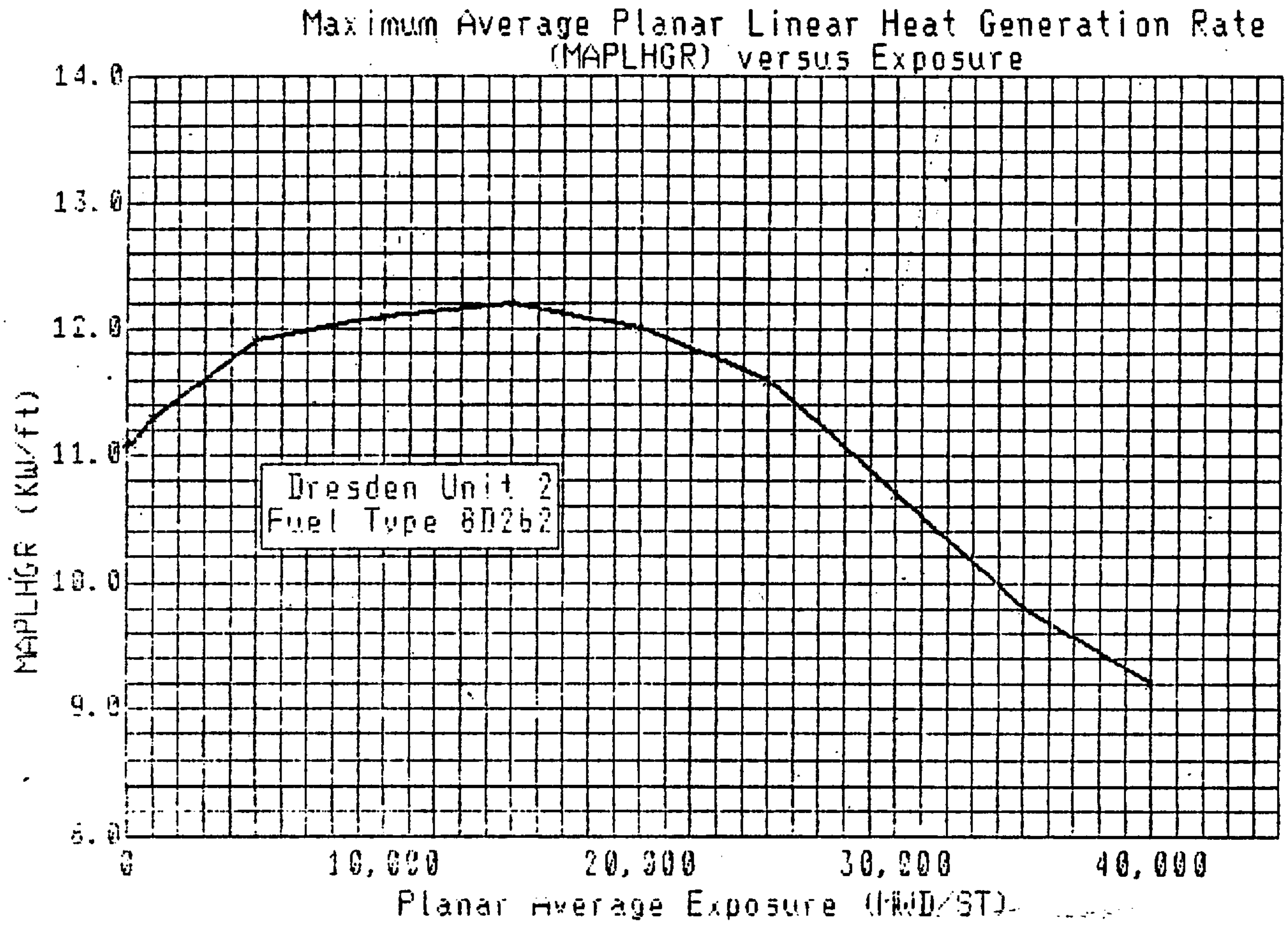
The LHGR shall be checked daily during reactor operation at greater than or equal to 25% rated thermal power.



The above graph is based on the following MAPLHGR summary for fuel type 8D250.

Planar Average Exposure (MWD/ST)	MAPLHGR (KW/Ft)
200	11.2
1,000	11.3
5,000	11.9
10,000	12.1
15,000	12.2
20,000	12.0
25,000	11.5
30,000	10.6
35,000	9.6
40,000	9.0

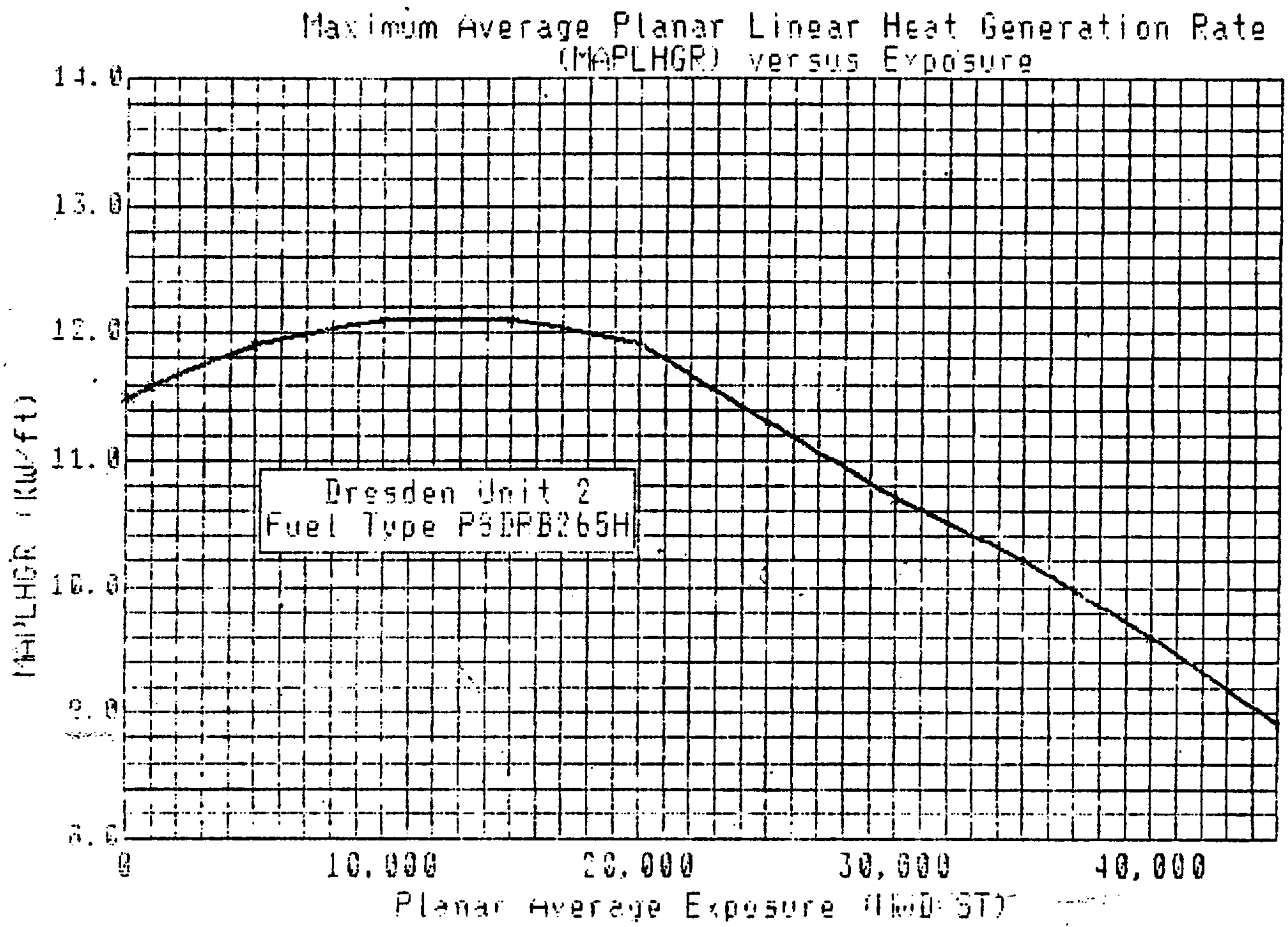
Figure 3.5-1
(Sheet 1 of 8)



The above graph is based on the following MAPLHGR summary for fuel type 8D262.

Planar Average Exposure (MWD/ST)	MAPLHGR (KW/Ft)
200	11.1
1,000	11.3
5,000	11.9
10,000	12.1
15,000	12.2
20,000	12.0
25,000	11.6
30,000	10.7
35,000	9.8
40,000	9.2

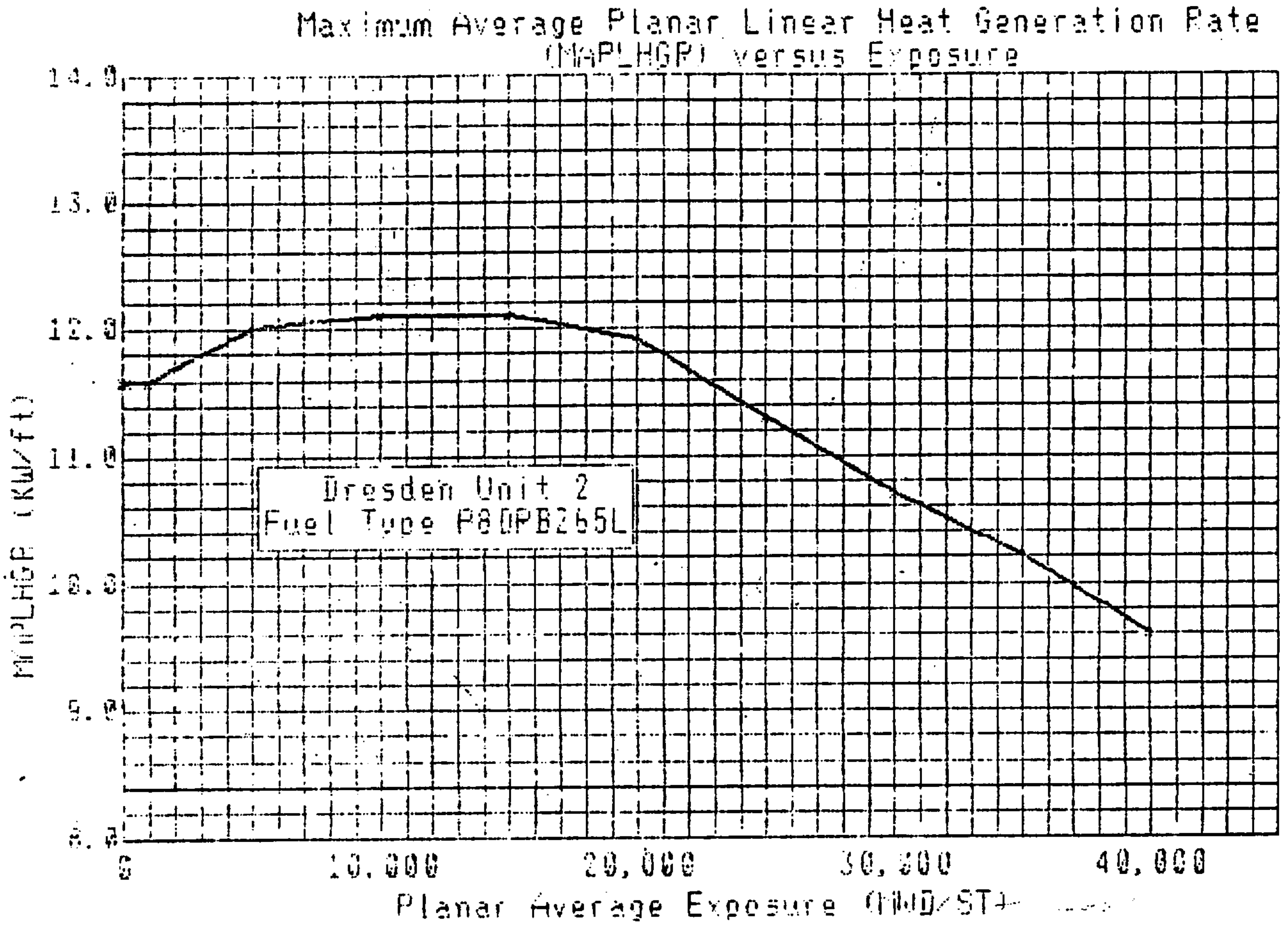
Figure 3.5-1
(Sheet 2 of 8)



The above graph is based on the following MAPLHGR summary for fuel type P8DRB265H.

Planar Average Exposure (MWD/ST)	MAPLHGR (KW/Ft)
200	11.5
1,000	11.6
5,000	11.9
10,000	12.1
15,000	12.1
20,000	11.9
25,000	11.3
30,000	10.7
35,000	10.2
40,000	9.6
45,000	8.9

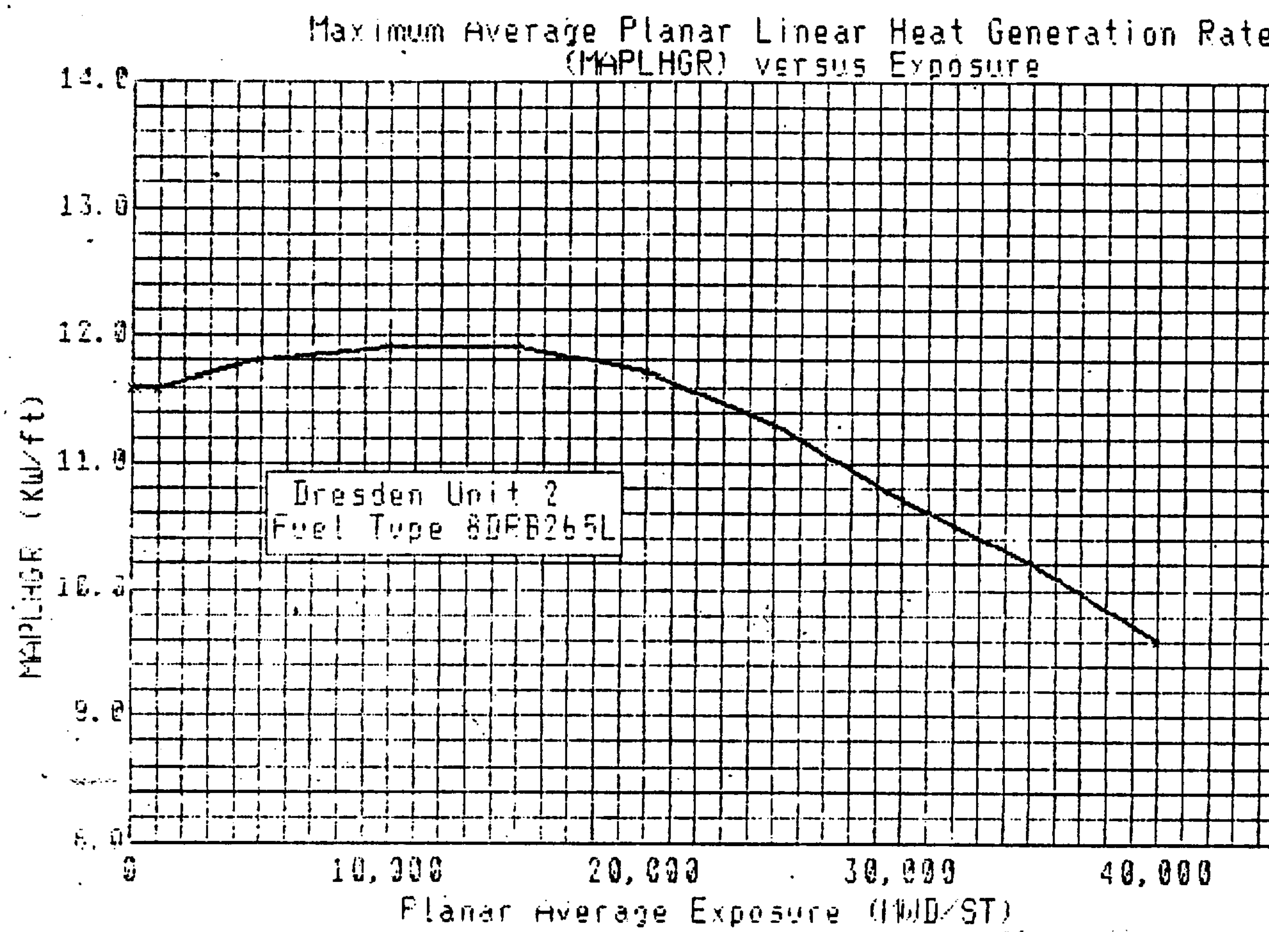
Figure 3.5-1
(Sheet 3 of 8)



The above graph is based on the following MAPLHGR summary for fuel type P8DRB265L.

Planar Average Exposure (MWD/ST)	MAPLHGR (KW/Ft)
200	11.6
1,000	11.6
5,000	12.0
10,000	12.1
15,000	12.1
20,000	11.9
25,000	11.3
30,000	10.7
35,000	10.2
40,000	9.6

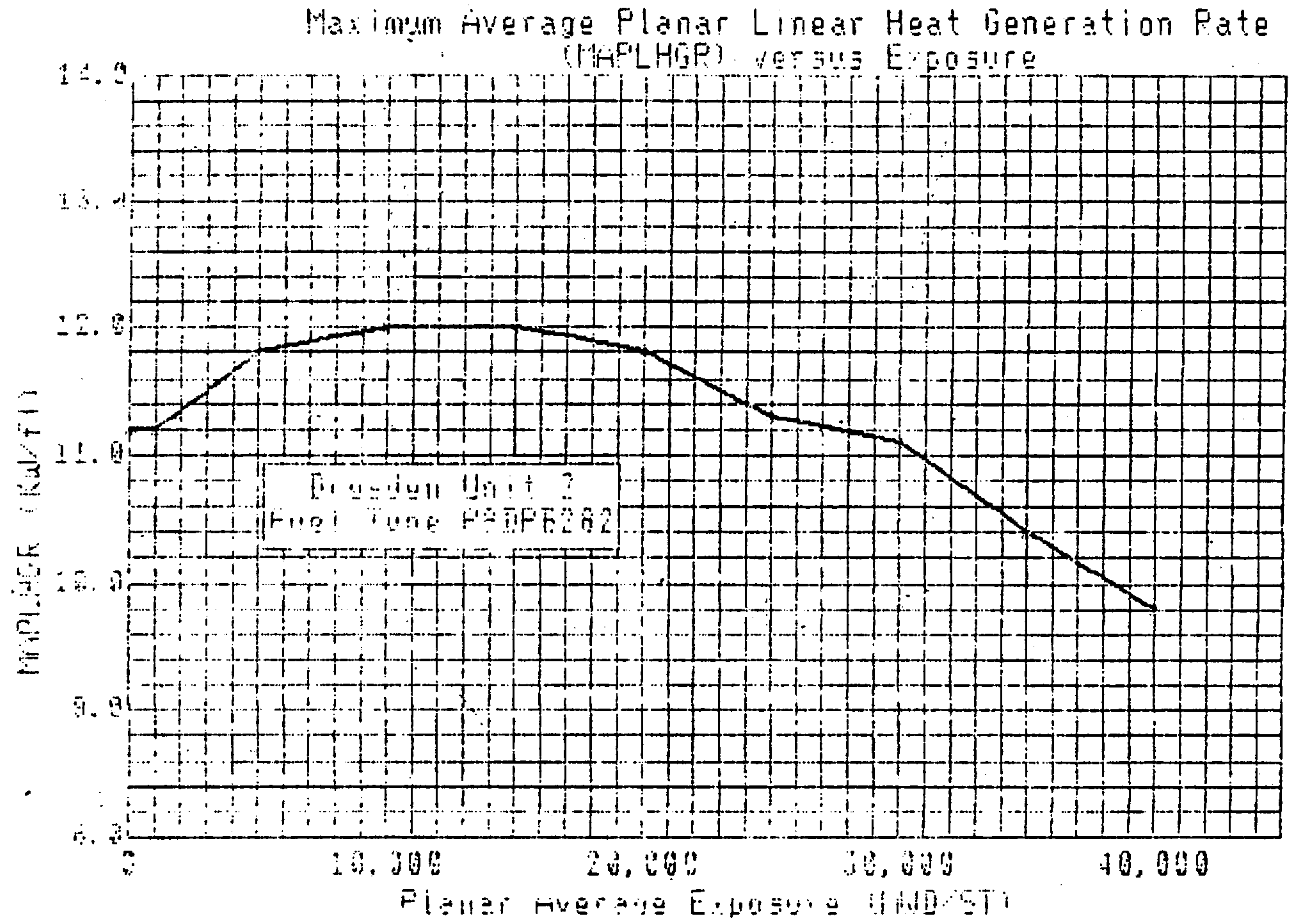
Figure 3.5-1
(Sheet 4 of 8)



The above graph is based on the following MAPLHGR summary for fuel type 8DRB265L.

Planar Average Exposure (MWD/ST)	MAPLHGR (KW/Ft)
200	11.6
1,000	11.6
5,000	11.8
10,000	11.9
15,000	11.9
20,000	11.7
25,000	11.3
30,000	10.7
35,000	10.2
40,000	9.6

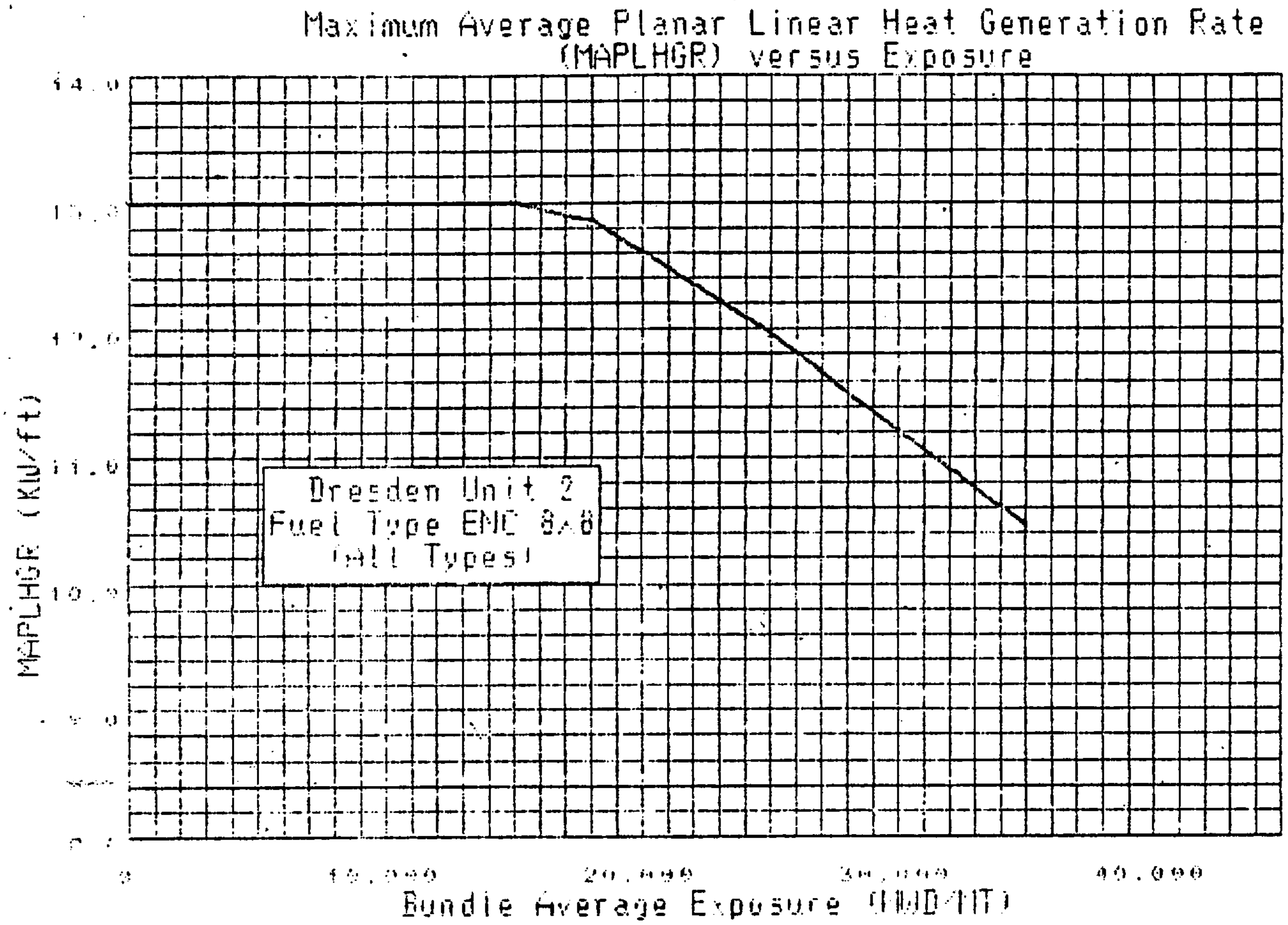
Figure 3.5-1
(Sheet 5 of 8)



The above graph is based on the following MAPLHGR summary for fuel type P8DRB282.

Planar Average Exposure (MWD/ST)	MAPLHGR (KW/Ft)
200	11.2
1,000	11.2
5,000	11.8
10,000	12.0
15,000	12.0
20,000	11.8
25,000	11.3
30,000	11.1
35,000	10.4
40,000	9.8

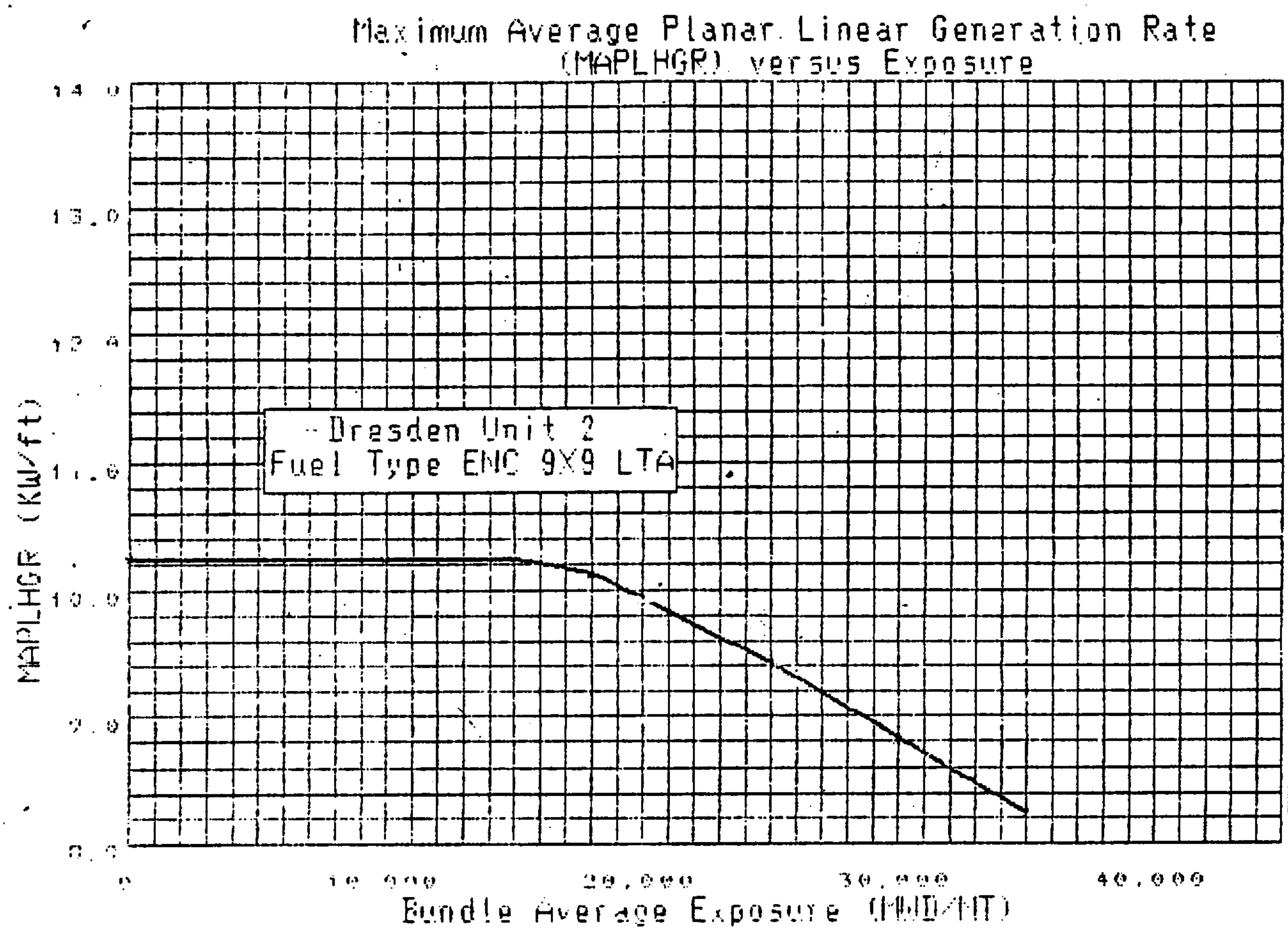
Figure 3.5-1
(Sheet 6 of 8)



The above graph is based on the following MAPLHGR summary for fuel type ENC 8X8 (all types).

Bundle Average Exposure (MWD/MTU)	MAPLHGR (KW/Ft)
0	13.0
15,000	13.0
18,000	12.85
20,000	12.60
25,000	11.95
30,000	11.20
35,000	10.45

Figure 3.5-1
(Sheet 7 of 8)



The above graph is based on the following MAPLHGR summary for fuel type ENC 9X9 LTA.

Bundle Average Exposure (MWD/MTU)	MAPLHGR (KW/Ft)
0	10.24
15,000	10.24
18,000	10.12
20,000	9.92
25,000	9.41
30,000	8.82
35,000	8.23

Figure 3.5-1
(Sheet 8 of 8)

3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

K. Minimum Critical Power Ratio (MCPR)

During steady state operation at rated core flow, MCPR shall be greater than or equal to; 1.34 for XN-1 8x8 and G.E. 8x8 Fuel types 1.35 for G.E. 8x8R 1.38 for XN-1 9x9 LTA

For core flows other than rated, the MCPR operating limit shall be as follows:

1. Manual Flow Control - the MCPR Operating Limit shall be the value from Figure 3.5-2 Sheet 1 or the above rated core flow value, which ever is greater.

2. Automatic Flow Control - the MCPR Operating Limit shall be the value from Figure 3.5-2 Sheet 1, Sheet 2 or the above rated core flow value, whichever is greatest.

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 REQUIREMENT (Cont'd.)

K. Minimum Critical Power Ratio (MCPR)

MCPR shall be determined daily during a reactor power operation at greater than or equal to 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 CONDITION FOR OPERATION (Cont'd.)

In the event the average 90% scram insertion time determined by Spec. 3.3.C for all operable control rods exceeds 2.74 seconds, the MCPR limit shall be increased by the amount equal to [0.092T - 0.252] where T equals the average 90% scram insertion time for the most recent half-core or full core surveillance data from Spec. 4.3.C.

L. Condensate Pump Room Flood Protection

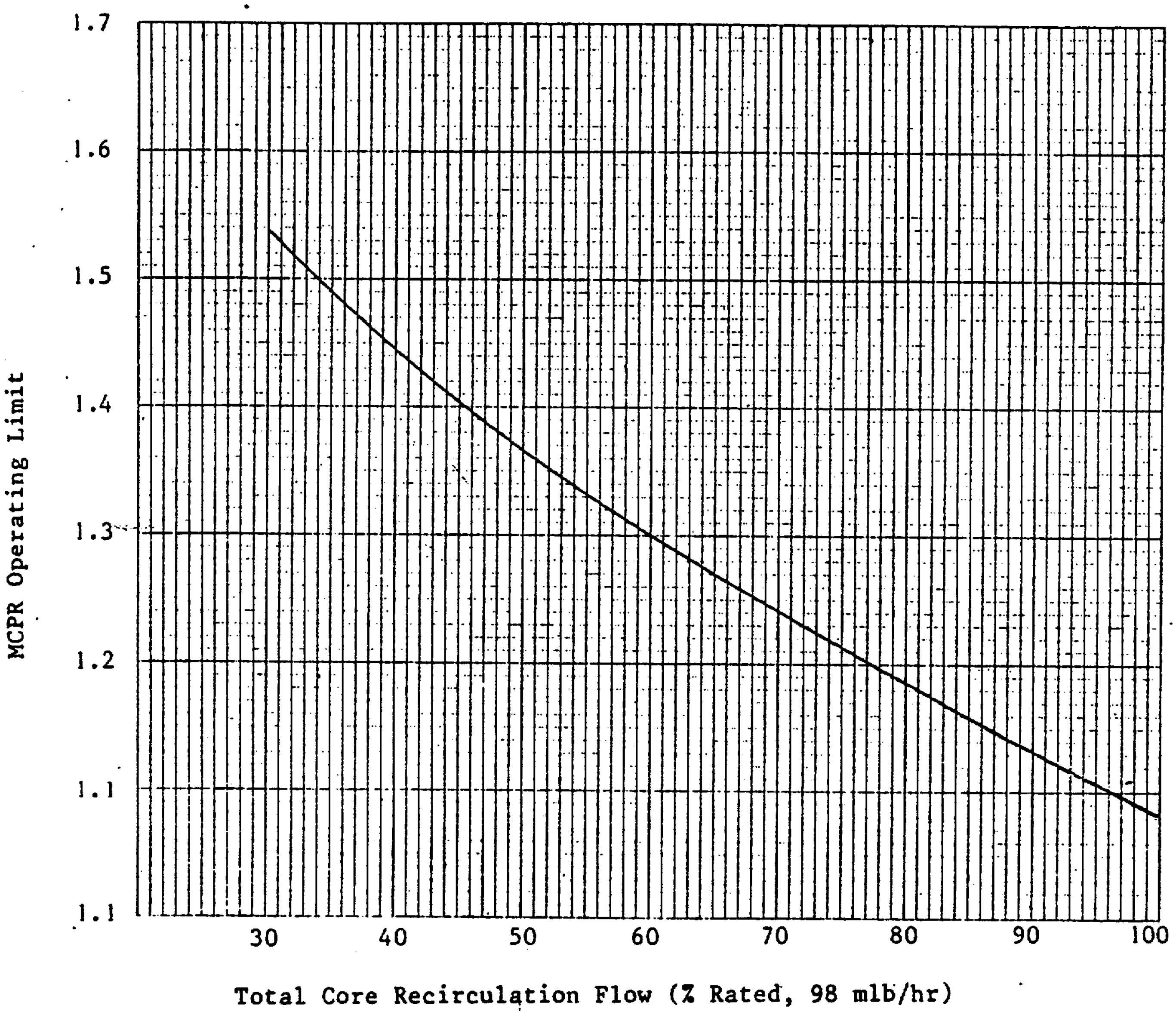
1. The system is installed to prevent or mitigate the consequences of flooding of the condensate pump room shall be operable prior to startup of the reactor.

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

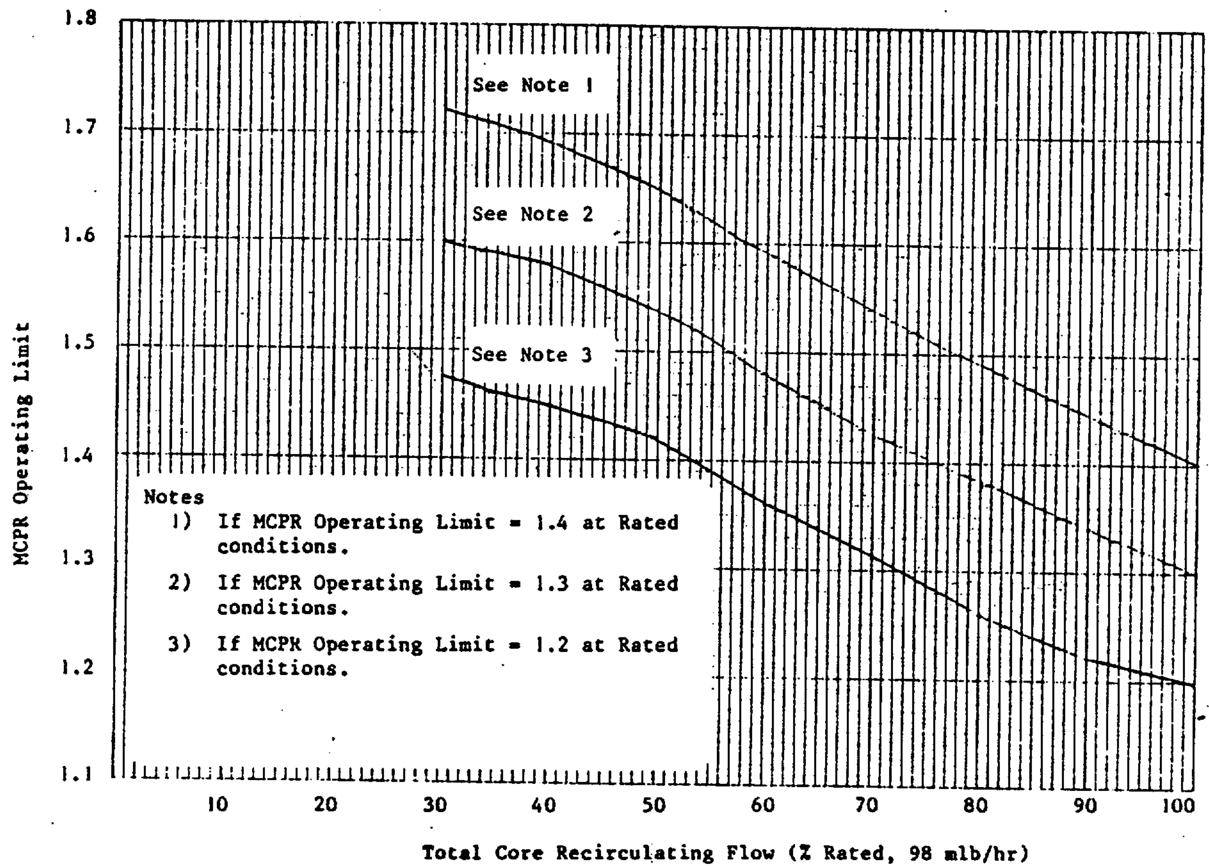
L. Condensate Pump Room Flood Protection

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

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



MCPR Limit For Reduced Core Flow
Figure 3.5-2
(Sheet 1 of 2)



MCPR Limit For Automatic Flow Control
Figure 3.5-2
(Sheet 2 of 2)

3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

b. The CCSW Vault Floor drain shall be checked during each operating cycle by assuring that water can be run through the drain line and actuating the air operated valves by operation of the following sensor:

   i. loss of air

   ii. high level in the condensate pump room (5'0")

c. The condenser pit five foot trip shall have a trip setting of less than or equal to five feet zero inches. The five foot trip circuit for each channel shall be checked once every three months. The 3 and 1 foot alarms shall have a setting of less than or equal to three feet zero inches and less than or equal to 1 foot 0 inches. A logic system functional test, including all alarms, shall be performed during the refueling outage.

3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

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

3. If Specification 3.5.L.1 and 2 cannot be met, reactor startup shall not commence or if operating, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours.

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

3.5 LIMITING CONDITION FOR OPERATION BASES

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

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

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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, Revisions 1, 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.)

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 judgement.

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

Should the loss of one LPCI pump occur, a nearly full complement of core and containment cooling equipment is available. Three LPCI pumps in conjunction with the core spray subsystem will perform the core cooling function. Because of the availability of the majority of the core cooling equipment, which will be demonstrated to be operable, a 30-day repair period is justified. If the LPCI subsystem is not available, at least 2 LPCI pumps must be available to fulfill the containment cooling function. The 7-day repair period is set on this basis.

B. Containment Cooling Service Water - The containment heat removal portion of the LPCI/containment cooling subsystem is provided to remove heat energy from the containment in the event of a loss of coolant accident. For the flow specified, the containment long-term pressure is limited to less than 8 psig and, therefore, is more than ample to provide the required heat removal capability. (Ref. Section 5.2.3.2 SAR).

The containment cooling subsystem consists of two sets of 2 service water pumps, 1 heat exchanger and 2 LPCI pumps. Either set of equipment is capable of performing the containment cooling function. Loss of one containment cooling service water pump does not seriously jeopardize the containment cooling capability as any 2 of the remaining three pumps can satisfy the cooling requirements. Since there is some redundancy left a 30-day repair period is adequate. Loss

3.5 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

of 1 containment cooling subsystem leaves one remaining system to perform the containment cooling function. The operable system is demonstrated to be operable each day when the above condition occurs. Based on the facts that when one containment cooling subsystem becomes inoperable only one system remains which is tested daily. A 7-day repair period was specified.

C. High Pressure Coolant Injection - The high pressure coolant injection subsystem is provided to adequately cool the core for all pipe breaks smaller than those for which the LPCI or core spray subsystems can protect the core.

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

D. Automatic Pressure Relief - The relief valves of the automatic pressure relief subsystem are a back-up to the HPCI subsystem. They enable the core spray or LPCI to provide protection against the small pipe break in the event of HPCI failure, by depressurizing the reactor vessel rapidly enough to actuate the core sprays or LPCI. The core spray and/or LPCI provide sufficient flow of coolant to adequately cool the core.

Loss of 1 of the relief valves affects the pressure relieving capability and therefore a 7 day repair period is specified. Loss of more than 1 relief valve significantly reduces the pressure relief capability and thus a 24-hour repair period is specified.

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

3.5 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

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

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

Dresden Units 2 and 3 share certain process systems such as the makeup demineralizers and the radwaste system and also some safety systems such as the standby gas treatment system, batteries, and diesel generators. All of these systems have been sized to perform their intended function considering the simultaneous operation of both units.

3.5 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

For the safety related shared features of each plant, the Technical Specifications for that unit contain the operability and surveillance requirements for the shared feature; thus, the level of operability for one unit is maintained independently of the status of the other. For example, the shared diesel (2/3 diesel) would be mentioned in the specifications for both Units 2 and 3 and even if Unit 3 were in the Cold Shutdown Condition and needed no diesel power, readiness of the 2/3 diesel would be required for continuing Unit 2 operation.

G. Specification 3.5.F.4 provides that should this occur, no work will be performed which could preclude adequate emergency cooling capability being available. Work is prohibited unless it is in accordance with specified procedures which limit the period that the control rod drive housing is open and assures that the worst possible loss of coolant resulting from the work will not result in uncovering the reactor core. Thus, this specification assures adequate core cooling. Specification 3.9 must be consulted to determine other requirements for the diesel generator.

Specification 3.5.F.5 provides assurance that an adequate supply of coolant water is immediately available to the low pressure core cooling systems and that the core will remain covered in the event of a loss of coolant accident while the reactor is depressurized with the head removed.

H. Maintenance of Filled Discharge Pipe - If the discharge piping of the core spray, LPCI, and HPCI are not filled, a water hammer can develop in this piping when the pump and/or pumps are started.

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 affects 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

3.5 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

local variations in power distribution within a fuel assembly affect the calculated peak clad temperature by less than plus or minus 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) and in reference (2). Power operation with APLHGRs 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.

The maximum average planar LHGRs for G.E. fuel 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 fuel rod fabricated by G.E. is less than the design linear heat generation rate even if fuel pellet densification is postulated.

For fuel fabricated by ENC, protection of the MCPR and MAPLHGR limits and operation within the power distribution assumptions of the Fuel Design Analysis provides adequate protection against cladding strain limits, hence the LHGR limitation for GE fuel is unnecessary for the protection of ENC fuel.

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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, Revision 1, April, 1979.

(2) XN-NF-82-88 "Dresden Unit 2 LOCA Analysis Using the ENC EXEM/BWR Evaluation Model MAPLHGR Results"

3.5 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

K. Minimum Critical Power Ratio (MCPR)

The steady-state values for MCPR specified in the Specification were determined using the THERMEX thermal limits methodology described in XN-NF-80-19, Volume 3. The safety limit implicit in the Operating limits is established so that during sustained operation at the MCPR safety limit, at least 99.9% of the fuel rods in the core are expected to avoid boiling transition. The Limiting Transient delta CPR implicit in the operating limits was calculated such that the occurrence of the limiting transient from the operating limit will not result in violation of the MCPR safety limit in at least 95% of the random statistical combinations of uncertainties.

Transient events of each type anticipated during operation of a BWR/3 were evaluated to determine which is most restrictive in terms of thermal margin requirements. The generator load rejection/turbine trip without bypass is typically the limiting event. The thermal margin effects of the event are evaluated with the THERMEX Methodology and appropriate MCPR limits consistent with the XN-3 critical power correlation are determined. Several factors influence which transient results in the largest reduction in critical power ratio, such as the cycle-specific fuel loading, exposure and fuel type. The current cycle's reload licensing analyses identifies the limiting transient for that cycle.

As described in Specification 4.3.C.3 and the associated Bases, observed plant data were used to determine the average scram performance used in the transient analyses for determining the MCPR Operating Limit. If the current cycle scram time performance falls outside of the distribution assumed in the analyses, an adjustment of the MCPR limit may be required to maintain margin to the MCPR Safety Limit during transients. Compliance with the assumed distribution and adjustment of the MCPR Operating Limit will be performed in accordance with Technical Specifications 4.3.C.3 and 3.5.K.

For core flows less than rated, the MCPR Operating Limit established in the specification is adjusted to provide protection of the MCPR Safety Limit in the event of an uncontrolled recirculation flow increase to the physical limit of pump flow. This protection is provided for manual and automatic flow control by choosing the MCPR operating limit as the value from Figure 3.5-2 Sheet 1 or the rated core flow value, whichever is greater. For Automatic Flow Control, in addition to protecting the MCPR Safety Limit during the flow

3.5 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

run-up event, protection is provided against violating the rated flow MCPR Operating Limit during an automatic flow increase to rated core flow. This protection is provided by the reduced flow MCPR limits shown in Figure 3.5-2 Sheet 2 where the curve corresponding to the current rated flow MCPR limit is used (linear interpolation between the MCPR limit lines depicted is permissible). Therefore, for Automatic Flow Control, the MCPR Operating Limit is chosen as the value from Figure 3.5-2 Sheet 1, Sheet 2 or the rated flow value, whichever is greatest. It should be noted that if the rated flow MCPR Limit must be increased due to degradation of control rod scram times during the current cycle, the new value of the rated flow MCPR limit is applied when using Figure 3.5-2 Sheet 2.

L. Flood Protection

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

4.5 SURVEILLANCE REQUIREMENT BASES

(A thru F)

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

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

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

The requirement of 180 psig at 3500 gpm at the containment cooling service water (CCSW) pump discharge provides adequate margin to ensure that the LPCI/CCSW system provides the design

4.5 SURVEILLANCE REQUIREMENT BASES (Cont'd.)

bases cooling water flow and maintains 20 psig differential pressure at the containment cooling heat exchanger. This differential pressure precludes reactor coolant from entering the river water side of the containment cooling heat exchangers.

The verification of Main Steam Relief Valve operability during manual actuation surveillance testing must be made independent of temperatures indicated by thermocouples downstream of the relief valves. It has been found that a temperature increase may result with the valve still closed. This is due to steam being vented through the valve actuation mechanism during the surveillance test. By first opening a turbine bypass valve, and then observing its closure response during relief valve actuation, positive verification can be made for the relief valve opening and passing steam flow. Closure response of the turbine control valves during relief valve manual actuation would likewise serve as an adequate verification for relief valve opening. This test method may be performed over a wide range of reactor pressure greater than 150 psig. Valve operation below 150 psig is limited by the spring tension exhibited by the relief valves.

G. Deleted

H. Maintenance of Filled Discharge Pipe

The surveillance requirements to assure that the discharge piping of the core spray, LPCI, and HPCI systems are filled provides for a visual observation that water flows from a high point vent. This ensures that the line is in a full condition. Between the monthly intervals at which the lines are vented, instrumentation has been provided to monitor the presence of water in the discharge piping. This instrumentation will be calibrated on the same frequency as the safety system instrumentation. This period of periodic testing ensures that during the intervals between the monthly checks the status of the discharge piping is monitored on a continuous basis.

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

4.5 SURVEILLANCE REQUIREMENT BASES (Cont'd.)

margin; therefore, evaluation of the average planar LHGR below this power level is not necessary. The daily requirement for calculating average planar LHGR above 25 per cent rated thermal power is sufficient since power distribution shifts are slow when there have not been significant power or control rod changes.

J. Local LHGR

The LHGR for G.E. fuel 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 permissible 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 requirements by a considerable margin. With this low void content, any inadvertent core flow increase would only place operation in a more conservative mode relative to MCPR.

The daily requirement for calculating MCPR above 25 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.

L. Flood Protection

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

4.5 SURVEILLANCE REQUIREMENT BASES (Cont'd.)

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

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

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

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

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

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

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

a. Loss of air or power

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

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

4.5 SURVEILLANCE REQUIREMENT BASES (Cont'd.)

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

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

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

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

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

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

4.5 SURVEILLANCE REQUIREMENT BASES (Cont'd.)

for the heat supplied to the vault by the 2B pump motor keeping the vault at less than 105°F.

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

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

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

5.0 DESIGN FEATURES

5.1 Site

Dresden Unit 2 is located at the Dresden Nuclear Power Station which consists of a tract of land of approximately 953 acres located in the northeast quarter of the Morris 15-minute quadrangle (as designated by the United States Geological Survey), Goose Lake Township, Grudy County, Illinois. The tract is situated in portions of Sections 25, 26, 27, 34, 35, and 36 of Township 34 North, Range 8 East of the Third Principal Meridian.

5.2 Reactor

A. The core shall consist of not more than 724 fuel assemblies

B. The reactor core shall contain 177 cruciform-shaped control rods. The control material shall be boron carbide powder (B_4C) compacted to approximately 70% of theoretical density, or Hafnium metal.

5.3 Reactor Vessel

The reactor vessel shall be as described in Table 4.1.1 of the SAR. The applicable design codes shall be as described in Table 4.1.1 of the SAR.

5.4 Containment

A. The principal design parameters and applicable design codes for the primary containment shall be as given in Table 5.2.1 of the SAR.

B. The secondary containment shall be as described in Section 5.3.2 of the SAR and the applicable codes shall be as described in Section 12.1.1.3 of the SAR.

C. Penetrations to the primary containment and piping passing through such penetrations shall be designed in accordance with standards set forth in Section 5.2.2 of the SAR and the applicable codes shall be as described in Section 12.1.1.3 of the SAR.

5.5 Fuel Storage

A. The new fuel storage facility shall be such that the K_{eff} dry is less than 0.90 and flooded is less than 0.95.

B. The K_{eff} of the spent fuel storage pool shall be less than or equal to 0.95.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 84 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 letters dated September 11, 27 and 28, 1984 and October 2, 1984 (References 1-4), Commonwealth Edison Company (CECo) (the licensee) proposed to amend Appendix A of Provisional Operating License No. DPR-19. The letters furnished information to support authorization for Dresden 2 to operate during Cycle 10 with reload fuel supplied by and the associated analyses performed by Exxon Nuclear Company and to allow for the use of hafnium as a control rod absorber material. For Cycle 10, the licensee plans to use General Electric (GE) hybrid design hafnium control rod blades in Dresden 2. The letters also provided information to support new limiting conditions for operation and surveillance requirements for a newly modified scram system having improved reliability and changes in the calibration and functional test frequencies for certain specific instruments that are being modified into analog trip systems.

Specifically related to the reload fuel, the licensee requested extension of the maximum average planar linear heat generation rate (MAPLHGR) curves for 8 x 8 and 9 x 9 (Lead Test Assembly) (LTA) fuel types and for GE P8DRB265H fuel type and deletion of the MAPLHGR curve for GE fuel type P8DRB239 which has never been used at Dresden and is not expected to be in the future.

Notices of Consideration of Issuance of Amendment to License and Proposed No Significant Hazards Consideration Determination and Opportunity for Hearing related to the requested actions in the aforementioned letters were published in the Federal Register on October 24, 1984 (49 FR 42815) and November 21, 1984 (49 FR 45944 and 45945). No request for hearing and no comments were received.

2.0 EVALUATION OF THE PROPOSED EXTENSION OF THE BURNUP RANGE FOR MAPLHGR CURVES DURING D2 CYCLE 10

2.1 Background

Currently the MAPLHGR Technical Specifications for Exxon 8 x 8 and 9 x 9 LTA fuel assemblies extend only to a burnup of 13,500 MWD/MTU. During Cycle 10 both of these fuel types are expected to exceed that limit.

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The licensee is also proposing to change the GE P8DRB265H fuel type burnup limit of the MAPLHGR curve to 45,000 MWD/ST. Although this fuel type is not expected to reach this burnup during Cycle 10, the licensee is proposing a change now to preclude a separate submittal and NRC review at a later date.

2.2 Evaluation

The staff has reviewed not only the proposed changes to the MAPLHGR curves themselves but the other aspects of fuel safety which will be affected by the proposed increase in burnup.

The staff is currently reviewing extended burnup topical reports for both GE and Exxon (References 5 and 6). The review of these reports served as the basis for the extension in burnup proposed by the licensee. Although the review of these reports is not yet complete, the review has progressed to the point where the staff can find an acceptable basis for the burnup extension proposed by the licensee. Only one significant issue was found to affect the proposed burnup extension. This was the selection of a power history for the calculation to demonstrate that the fuel rod internal pressure remains less than the reactor coolant system pressure for the Exxon fuel. Since generic discussions are continuing with Exxon as part of the review of Reference 6, the staff performed an acceptable, independent fuel rod pressure calculation (with the assistance of Battelle Pacific Northwest Laboratories) and determined that the internal pressure of the Exxon fuel rods will not exceed the reactor coolant system pressure in Dresden 2.

In order to determine the MAPLHGR limits, approved calculational methods were used by the licensee for both the Exxon and GE fuel assemblies (References 7 through 10). For the Exxon 9 x 9 LTAs, the licensee obtained the MAPLHGR curve from the Exxon 8 x 8 fuel assemblies by conserving the allowable nodal powers and ratioing the active fuel lengths. This method was used to generate the MAPLHGR curves currently in the Dresden 2 Technical Specifications for these 9 x 9 LTAs. Since it is not obvious that this method is conservative, the licensee performed several loss-of-coolant accident (LOCA) heatup calculations to demonstrate that the peak clad temperature of these 9 x 9 LTAs would remain below the LOCA criteria of 10 CFR 50.46 for the existing MAPLHGR curves (Reference 11). Since the stored energy is highest between zero and 13,500 MWD/MTU these calculations are sufficient to cover the longer time period. The GE fuel assemblies have already been approved for operation to 40,000 MWD/ST. The licensee is proposing an extension to 45,000 MWD/ST. The staff finds this acceptable since the staff has generically approved an extension in burnup for this fuel type to 50,000 MWD/ST in GESTAR-II (Reference 12) as long as approved methods are used for the LOCA analysis, as is the case for this proposed change to the Technical Specifications.

2.3 Summary

Based on the analysis above, the staff has concluded that the requested extensions of the MAPLHGR curves are acceptable. This is based on the following considerations:

1. The staff independently determined that the internal pressure of the Exxon fuel rods will not exceed the reactor coolant system pressure during D2 Cycle 10.

2. Calculational methods approved by the staff were used to determine the MAPLHGR limits requested by the licensee.

3.0 USE OF HAFNIUM AS A CONTROL ROD ABSORBER MATERIAL FOR DRESDEN 2

3.1 Background

At present, the Dresden 2 Technical Specifications permit only B_4C to be used as control rod absorber material although the use of both B_4C and hafnium has been approved for use in Dresden 3 (Reference 13). The licensee's October 2, 1984 letter requested that hafnium also be permitted for control rod absorber material in Dresden 2 and proposed that GE type Hybrid I Control Rod (GE Type I HICR) assemblies which were discussed in the GE Topical Report NEDE-22290-A (Safety Evaluation of the General Electric Hybrid I Control Rod Assembly, September 1983, GE proprietary) be installed.

3.2 Evaluation and Conclusion

The staff has previously approved, on a plant-specific basis, the use of hafnium as a control rod absorber material in ASEA-ATOM control rods inserted in Dresden 3 (Reference 13). While CECo is proposing that the GE Type I HICR assemblies be installed in Dresden 2 during this outage, the amendment request is for the general use of hafnium as a control rod absorber material. As indicated above, its use in ASEA-ATOM blades has been approved by the staff for use in its identical sister unit, Dresden 3, and its use in the GE Type I HICR assemblies described in GE Topical Report NEDE-22290-A has been reviewed and approved by the staff in a Safety Evaluation letter dated August 22, 1983. Both control rod designs have the same worth and weight as the existing blades. The differences in design are in the cladding and absorber material and serve to improve blade lifetime. The staff, therefore, concludes that the use of hafnium in control rod blades for Dresden 2 is acceptable providing that the previously approved designs are installed.

4.0 SCRAM DISCHARGE SYSTEM

A Generic Safety Evaluation for the modified scram discharge system, issued December 10, 1980, endorsed the criteria set forth by the BWR Owners subgroup to meet the concerns arising from the Browns Ferry incomplete scram event of July 1980. By the NRC Confirmatory Order of June 24, 1982, the licensee's commitment to modify its scram discharge system in response to these concerns was confirmed (Reference 14). Also, model Technical Specifications were forwarded to the licensee as guidance for revising the Technical Specifications for operation with the newly modified scram discharge system. Following a period of discussion with the licensee regarding the application of the model Technical Specifications to the unit-specific Technical Specifications for the Dresden Nuclear Power Station, the licensee, by letter dated September 27, 1984, proposed Technical Specifications for the newly modified scram discharge system for Unit 2. The Technical Specification changes proposed in the licensee's submittal are fully responsive to the concerns addressed in the Generic Safety Evaluation on Scram Discharge Systems and are in keeping with the guidance provided in the model Technical Specifications. Therefore, the staff finds the proposed changes acceptable.

5.0 ANALOG TRIP INSTRUMENT SURVEILLANCE FREQUENCY

Certain equipment is being replaced to satisfy the requirement of 10 CFR 50.49 regarding environmental qualification of electrical equipment important to safety. In association with these changes, several existing instruments will be converted into analog trip systems; these are:

Reactor Low Water Level Instrument, 2-263-57A and B and 2-263-58A and B
Reactor Water High Level Instrument, 2-263-73A and B
HPCI High Steam Flow Instrument, 2-2389A thru D
HPCI Steam Line Low Pressure Instrument, 2-2352 and 2353

The analog trip systems consist of an analog sensor and transmitter, and a trip unit arrangement which ultimately actuates a trip relay. The frequency of calibration and functional testing for instrument loops of the analog trip system has been established in Reference 15, an NRC-approved reference document. With the currently installed one-out-of-two, taken twice logic, the prescribed calibration/functional test frequency for the respective transmitters, however, is once per operating cycle. The Technical Specification changes proposed in Reference 3 would require the channel calibration to be performed at the transmitter at a frequency of once per operating cycle. Since this is the calibration frequency recommended in the NRC-approved GE Topical Report, NEDO-21617-A (Reference 15), and the proposal is to conform the surveillance requirements to the recommended and NRC-approved period, the licensee's proposal is acceptable.

6.0 ENVIRONMENTAL CONSIDERATION

The amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes to the surveillance requirements. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

7.0 CONCLUSIONS

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

8.0 ACKNOWLEDGEMENT

The following staff members have contributed to this evaluation: A. Gill and R. Gilbert.

Dated: January 17, 1985

REFERENCES

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6. XN-NF-82-06-P, Rev. 1, Qualification of Exxon Nuclear Fuel For Extended Burnup, June 1982. (Proprietary)
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10. NEDO-24146A, "Loss-of-Coolant Accident Analysis Report for Dresden Units 2, 3 and Quad Cities Units 1, 2 Nuclear Power Stations," Revision 1, April 1979 as subsequently modified by Errata and Addenda 1 through 12.
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