February 8, 1990

Docket Nos. 50-237
and 50-249

Mr. Thomas J. Kovach
Nuclear Licensing Manager
Commonwealth Edison Company-Suite 300
OPUS West III
1400 OPUS Place
Downers Grove, Illinois 60515

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GPA/PA
OC/LFMB
DHagan
[AMENDMENT 75215/16]

Dear Mr. Kovach:

SUBJECT: TECHNICAL SPECIFICATION AMENDMENT AND CORE OPERATING LIMITS REPORT PER GENERIC LETTER 88-16 (TAC NOS. 75215 AND 75216)

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment Nos. 110 and 105 to Appendix A, Technical Specifications, of Provisional and Facility Operating License Nos. DPR-19 and DPR-25 for the Dresden Nuclear Power Station (DNPS), Units 2 and 3 respectively. These amendments are in response to an application from Commonwealth Edison Company dated October 30, 1989, as supplemented on November 20, 1989.

These amendments revise DNPS Technical Specifications, in accordance with Generic Letter 88-16, by replacing cycle-specific operating limits with references to a Core Operating Limits Report.

A copy of our related Safety Evaluation is also enclosed. The Notice of Issuance for these amendments will be included in the Commission's biweekly Federal Register notices.

Sincerely,

/S/

Byron Siegel, Project Manager
Project Directorate III-2
Division of Reactor Projects - III,
IV, V and Special Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 110 to License No. DPR-19
2. Amendment No. 105 to License No. DPR-25
3. Safety Evaluation

cc w/enclosures:
See next page

*See previous concurrence

*PD32:LA	PD32:PM	*PD32:PD	*OGC
LLUTHER	BSIEGEL/bj	JCRAIG	MYOUNG
01/22/90	1/16/90	01/25/90	01/31/90

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PDR ADOCK 05000237
P PDC

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PD32:DA	PD32:PM	PD32:PD	OGC
LLUTHER	BSIEGEL/bj	JCRAIG	
1/30/90	1/23/90	1/26/90	1/31/90

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

February 8, 1990

Docket Nos. 50-237
and 50-249

Mr. Thomas J. Kovach
Nuclear Licensing Manager
Commonwealth Edison Company-Suite 300
OPUS West III
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Byron Siegel, Project Manager
Project Directorate III-2
Division of Reactor Projects - III,
IV, V and Special Projects
Office of Nuclear Reactor Regulation

Enclosures:
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2. Amendment No. 105 to License No. DPR-25
3. Safety Evaluation

cc w/enclosures:
See next page

Mr. Thomas J. Kovach
Commonwealth Edison Company

Dresden Nuclear Power Station
Units 2 and 3

cc:

Michael I. Miller, Esq.
Sidley and Austin
One First National Plaza
Chicago, Illinois 60690

Mr. J. Eenigenburg
Plant Superintendent
Dresden Nuclear Power Station
Rural Route #1
Morris, Illinois 60450

U. S. Nuclear Regulatory Commission
Resident Inspectors Office
Dresden Station
Rural Route #1
Morris, Illinois 60450

Chairman
Board of Supervisors of
Grundy County
Grundy County Courthouse
Morris, Illinois 60450

Regional Administrator
Nuclear Regulatory Commission, Region III
799 Roosevelt Road, Bldg. #4
Glen Ellyn, Illinois 60137

Illinois Department of Nuclear Safety
Office of Nuclear Facility Safety
1035 Outer Park Drive
Springfield, Illinois 62704

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. 110
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 July 11, 1989, as supplemented by August 14, 1989, 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. of Provisional Operating License No. DPR-19 is hereby amended to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 110, 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 to be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

John W. Craig, Director
Project Directorate III-2
Division of Reactor Projects - III,
IV, V and Special Projects
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 8, 1990

ATTACHMENT TO LICENSE AMENDMENT NO. 110

PROVISIONAL OPERATING LICENSE DPR-19

DOCKET NO. 50-237

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change.

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1.0 DEFINITIONS

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

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

B. Core Operating Limits Report (COLR) - The Core Operating Limits Report is the unit specific document that provides core operating limits for the current operating reload cycle. These cycle specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.6.A.4. Plant operation within these operating limits is addressed in individual specifications.

C. Critical Power Ratio (CPR) - The critical power ratio is the ratio of that assembly power which causes some point in the assembly to experience transition boiling to the assembly power at the reactor condition of interest as calculated by application of the ANF NRC-approved correlation.

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

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

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

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

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

I. Limiting Conditions for Operation (LCO) - The limiting conditions for operation specify the minimum acceptable levels of system performance necessary to assure safe startup and operation of the facility. When these conditions are met, the

1.0 Definitions (Continued)

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

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

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

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

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

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

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

DD. 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. Fuel Design Limiting Ratio (FDLRX) - The fuel design limiting ratio is the limit used to assure that the fuel operates within the end-of-life steady state design criteria. FDRLX assures acceptable end-of-life conditions by, among other items, limiting the release of fission gas to the cladding plenum.

GG. Dose Equivalent I-131 - That concentration of I-131 (microcurie/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites".

1.1 SAFETY LIMIT BASES

FUEL CLADDING INTEGRITY

The fuel cladding integrity limit is set such that no calculated fuel damages 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 greater than the MCPR fuel cladding integrity safety limit represents a conservative margin relative to the conditions required to maintain fuel cladding integrity by assuring that the fuel does not experience transition boiling.

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 corrosions 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 perforation signals 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. The MCPR fuel cladding integrity Safety Limit assures that during normal operation and during anticipated operational occurrences, at least 99.9% of the fuel rods in the core do not experience transition boiling.

A. Reactor Pressure greater than 800 psig and Core Flow greater than 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

1.1 SAFETY LIMIT BASES (Cont'd.)

power ratio (CPR) which is the ratio of the bundle power which would produce the onset of transition boiling divided by the actual bundle power. The minimum value of this ratio for any bundle in the core is the Minimum Critical Power Ratio (MCPR). It is assumed that the plant operation is controlled to the nominal protective setpoints via the instrumented variables. (Figure 2.1-3).

The MCPR Fuel Cladding Integrity Safety Limit assures sufficient conservatism in the operating MCPR limit that in the event of an anticipated operational occurrence from the limiting condition for operation, at least 99.9% of the fuel rods in the core would be expected to avoid boiling transition. The margin between calculated boiling transition (MCPR=1.00) and the MCPR Fuel Cladding Integrity Safety Limit is based on a detailed statistical procedure which considers the uncertainties in monitoring the core operating state. One specific uncertainty included in the safety limit is the uncertainty inherent in the ANF NRC-approved critical power correlation. Refer to Specification 6.6.A.4 for the methodology used in determining the MCPR Fuel Cladding Integrity Safety Limit.

The ANF NRC-approved critical power correlation is based on a significant body of practical test data, providing a high degree of assurance that the critical power as evaluated by the correlation is within a small percentage of the actual critical power being estimated. The assumed reactor conditions used in defining the safety limit introduce conservatism into the limit because boundingly high radial power peaking factors and boundingly flat local peaking distributions are used to estimate the number of rods in boiling transition. Still further conservatism is induced by the tendency of the ANF NRC-approved correlation to overpredict the number of rods in boiling transition. These conservatisms and the inherent accuracy of the ANF NRC-approved correlation provide a reasonable degree of assurance that during sustained operation at the MCPR Fuel Cladding Integrity Safety Limit there would be no transition boiling in the core. If boiling transition were to occur, however, there is reason to believe that the integrity of the fuel would not necessarily be compromised. Significant test data accumulated by the U.S. Nuclear Regulatory Commission and private organizations indicate that the use of a boiling transition limitation to protect against cladding failure is a very conservative approach; much of the data indicates that LWR fuel can survive for an extended period in an environment of transition boiling.

During Single Loop Operation, the MCPR safety limit is increased by 0.01 to conservatively account for increased uncertainties in the core flow and TIP measurements.

1.1 SAFETY LIMIT BASES (Cont'd.)

If the reactor pressure should ever exceed the limit of applicability of the ANF NRC-approved critical power correlation as defined in the ANF NRC-approved methodology listed in Specification 6.6.A.4, it would be assumed that the MCPR Fuel Cladding Integrity Safety Limit had been violated. This applicability pressure limit is higher than the pressure safety limit specified in Specification 1.2.

Fuel design criteria have been established to provide protection against fuel centerline melting and 1% plastic cladding strain during transient overpower conditions throughout the life of the fuel. To demonstrate compliance with these criteria, fuel rod centerline temperatures are determined at 120% overpower conditions as a check against calculated centerline melt temperatures. FDLRC is incorporated to protect the above criteria at all power levels considering events which cause the reactor power to increase to 120% of rated thermal power.

B. Core Thermal Power Limit (Reactor Pressure less than 800 psia)

At pressures below 800 psia, the core elevation pressure drop (0 power, 0 flow) is greater than 4.56 psi. At low powers and flows this pressure differential is maintained in the bypass region of the core. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low powers and flows will always be greater than 4.56 psi. Analyses show that with a flow of 28×10^3 lbs/hr. bundle flow, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow

1.1 SAFETY LIMIT BASES (Cont'd.)

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. See the ANF NRC-approved methodology listed in Specification 6.6.A.4.

Conservatism is incorporated into the transient analyses which define the MCPR operating limits. Variables which inherently possess little or no uncertainty or whose uncertainty has little or no effect on the outcome of the limiting transient are selected at bounding values. Variables which possess significant uncertainty that may have undesirable effects on thermal margins are addressed statistically. Statistical methods used in the transient analyses are described in the ANF NRC-approved methodology listed in Specification 6.6.A.4. The MCPR operating limits are established such that the occurrence of the limiting transient will not result in the violation of the MCPR Fuel Cladding Integrity Safety Limit in at least 95% of the random statistical combinations of uncertainties. In general, the variables with the greatest statistical significance to the consequences of anticipated operational occurrences are the reactivity feedback associated with the formation and removal of coolant voids and the timing of the control rod scram.

2.1 LIMITING SAFETY SYSTEM SETTING BASES (Cont'd.)

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.L 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 during dual loop operation or 116.5 percent during single loop operation, 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.

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.

2.2 LIMITING SAFETY SYSTEM SETTING BASES

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 direct valve position trip scrams and 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 ANF NRC-approved methodology as specified in Specification 6.6.A.4 for each reload to assure that the pressure safety limit is not exceeded.

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)	
	Dual Loop Operation	Less than or equal to (.58 W_D plus 50)/FDLRC (See Note 2)
	Single Loop Operation	Less than or equal to (.58 W_D plus 46.5)/FDLRC (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)	
	Dual Loop Operation	See Core Operating Limits Report
	Single Loop Operation	See Core Operating Limits Report
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)

3.3 LIMITING CONDITION FOR OPERATION (Cont'd.)

2. The maximum scram insertion time for 90% insertion of any operable control rod shall not exceed 7.00 seconds.

D. Control Rod Accumulators

At all reactor operating pressures, a rod accumulator may be inoperable provided that no other control rod in the nine-rod square array around this rod has a:

1. Inoperable accumulator,

2. Directional control valve electrically disarmed while in a non-fully inserted position.

4.3 SURVEILLANCE REQUIREMENT (Cont'd.)

2. At 16 week intervals, at least 50% of the control rod drives shall be tested as in 4.3.C.1 so that every 32 weeks all of the control rods shall have been tested. Whenever 50% or more of the control rod drives have been tested, an evaluation shall be made to provide reasonable assurance that proper control rod drive performance is being maintained.

3. Following completion of each set of scram testing as described above, the results will be compared against the average scram speed distribution used in the transient analysis to verify the applicability of the current MCPR Operating Limit. Refer to Specification 3.5.L.

D. Control Rod Accumulators

Once a shift check the status of the pressure and level alarms for each accumulator.

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

3. The operability of the scram discharge volume vent and drain valves assures the proper venting and draining of the volume. This ensures that water accumulation does not occur which would cause an early termination of control rod movement during a full core scram. These specifications provide for the periodic verification that the valves are open and for testing of these valves under reactor scram conditions during each Refueling Outage.

B. Control Rod Withdrawal

1. Control rod dropout accidents as discussed in the ANF NRC-approved methodology listed in Specification 6.6.A.4, 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.

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 sub-critical 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

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

withdrawn could not be worth enough to cause the rod drop accident design limit of 280 cal/gm to be exceeded if they were to drop out of the core in the manner defined for the Rod Drop Accident. 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 RWM 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, 425 cal/gm, at which rapid fuel dispersal and primary system damage have been found to occur based on experimental data as is discussed in the ANF NRC-approved methodology listed in Specification 6.6.A.4.

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.

Parametric Control Rod Drop Accident analyses have shown that for wide ranges of key reactor parameters (which envelope the operating ranges of these variables), the fuel enthalpy rise during a postulated control rod drop accident remains considerably lower than the 280 cal/gm limit. For each operating cycle, cycle-specific parameters such as maximum control rod worth, Doppler coefficient effective delayed neutron fraction and maximum four-bundle local peaking factor are compared with the results of the parametric analyses to determine the peak fuel rod enthalpy rise. This value is then compared against the Technical Specification limit of 280 cal/gm to demonstrate compliance for each operating cycle. If cycle specific values of the above parameters are outside the range assumed in the parametric analyses, an extension of the analysis or a cycle specific analysis may be required. Conservatism present in the analysis, results of the parametric studies, and a detailed description of the methodology for performing the Control Rod Drop Accident analysis are provided in the ANF NRC-approved methodology listed in Specification 6.6.A.4.

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

analyses, and is also included in the allowable scram insertion times specified in Specification 3.3.C. The bounding value described above was used in the transient analysis.

The performance of the individual control rod drives is monitored to assure that scram performance is not degraded. Fifty percent of the control rod drives in the reactor are tested every sixteen weeks to verify adequate performance. Observed plant data or Technical Specification limits (Specification 3.3.C) were used to determine the average scram performance used in the transient analyses, and the results of each set of control rod scram tests performed per Specification 3.3.C during the current cycle are compared against earlier results to verify that the performance of the control rod insertion system has not changed significantly. If a test performed per Specification 3.3.C should be determined to fall outside of the statistical population defining the scram performance characteristics used in the transient analyses, a re-determination of thermal margin requirements is undertaken as required by Specification 3.5.L. A smaller test sample than that required by Specification 3.3.C is not statistically significant and should not be used in the re-determination of thermal margins. Control rod drives with excessive scram times can be fully inserted into the core and deenergized in the manner of an inoperable rod drive provided the allowable number of inoperable control rod drives is not exceeded. In this case, the scram speed of the drive shall not be used as a basis in the re-determination of thermal margin requirements.

The scram times for all control rods are measured at the time of each refueling outage. Experience with the plant has shown that control drive insertion times vary little through the operating cycle. 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 second is greater than 0.999 for a normal distribution.

D. Control Rod Accumulators

The basis for this specification was not described in the SAR and, therefore, is presented in its entirety. Requiring no more than one inoperable accumulator in any nine-rod square array is based on a series of XY PDQ-4 quarter core calculations of a cold, clean core. The worst case in a nine-rod withdrawal sequence resulted in a k_{eff} less than 1.0 -- other repeating rod sequences with more rods withdrawn resulted in k_{eff} greater than 1.0. At reactor pressures in excess of 800 psig, even those control rods with inoperable accumulators will be able to meet required scram insertion times due to the

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

action of reactor pressure. In addition, they may be normally inserted using the control-rod-drive hydraulic system. Procedural control will assure that control rods with inoperable accumulators will be spaced in a one-in-nine array rather than grouped together.

E. Reactivity Anomalies

During each fuel cycle excess operating reactivity varies as fuel depletes and as any burnable poison in supplementary control is burned. The magnitude of this excess reactivity may be inferred from the critical rod configuration. As fuel burnup progresses, anomalous behavior in the excess reactivity may be detected by comparison of the critical rod pattern selected base states to the predicted rod inventory at that state. Power operating base conditions provide the most sensitive and directly interpretable data relative to core reactivity. Furthermore, using power operating base conditions permits frequent reactivity comparisons. Requiring a reactivity comparison at the specified frequency assures that a comparison will be made before the core reactivity change exceeds 1% delta k. Deviations in core reactivity greater than 1% delta k are not expected and require thorough evaluation. One percent reactivity limit is considered safe since an insertion of the reactivity into the core would not lead to transients exceeding design conditions of the reactor system.

F. (N/A)

G. Economic Generation Control System

Operation of the facility with the Economic Generation Control System with automatic flow control is limited to the range of 65-100% of rated core flow. In this flow range and with reactor power above 20% the reactor can safely tolerate a rate of change of load of 8 MW(e)/sec. (Reference FSAR Amendment 9-Unit 2, 10-Unit 3). Limits within the Economic Generation Control System and Reactor Flow Control System preclude rates of change greater than approximately 4 MWe/sec.

When the Economic Generation Control System is in operation, this fact will be indicated on the main control room console. The results of initial testing will be provided to the NRC at the onset of routine operation with the Economic Generation Control System.

4.3 SURVEILLANCE REQUIREMENT BASES

None

3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

operation is permissible only during the succeeding seven(7) days provided that during such time the HPCI subsystem is operable. If the appropriate MAPLHGR reduction factors (multipliers) are applied to the MAPLHGR limits, the Automatic Pressure Relief Subsystem of ECCS shall be considered operable. The MAPLHGR Limits and the appropriate MAPLHGR reduction factors are found in the Core Operating Limits Report.

3. From and after the date that two relief valves are found or made to be inoperable, reactor operation is permissible only during the succeeding seven days provided that during such time the HPCI subsystem is operable and the multipliers specified in 3.5.D.2 are applied.

4. If the requirements of 3.5.D.1 cannot be met, an orderly shutdown shall be initiated and the reactor pressure shall be reduced to below 150 psig within 24 hours.

E. Isolation Condenser System

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

E. Surveillance of the Isolation Condenser System shall be performed as follows:

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 shall not exceed the MAPLHGR Limits. For Single Loop Operation (SLO), the MAPLHGR Limits shall be decreased by the SLO MAPLHGR Multiplicative factor(s). If concurrent with SLO, one Automatic Pressure Relief Subsystem Relief Valve is Out of Service (RVOOS), the MAPLHGR Limits shall be decreased by the RVOOS MAPLHGR Multiplicative factor(s). The MAPLHGR Limits and multiplicative factors are specified in the Core Operating Limits Report. 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.

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

I. Average Planar Linear Heat Generation Rate (APLHGR)

The APLHGR for each type of fuel shall be determined daily during reactor operation at greater than or equal to 25% rated thermal power.

3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

J. LOCAL STEADY STATE LHGR

During steady state power operation above 25% of rated thermal power, the linear heat generation rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed its maximum steady state LHGR (SLHGR) value shown in the Core Operating Limits Report. That is, the Fuel Design Limiting Ratio (FDLRX) shall not be greater than 1.0 where

$$FDLRX = \frac{LHGR}{SLHGR}$$

If at any time during operation above 25% rated thermal power, it is determined by normal surveillance that FDLRX for any fuel assembly exceeds 1.0, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the FDLRX 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.)

J. Linear Heat Generation Rate (LHGR)

The Fuel Design Limiting Ratio (FDLRX) shall be checked daily during reactor operation at greater than or equal to 25% rated thermal power.

3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

K. Local Transient LHGR

At any time during power operation, above 25% rated thermal power the fuel design limiting ratio for centerline melt (FDLRC) shall not be greater than 1.0, where

$$FDLRC = \frac{(LHGR)(1.2)}{(TLHGR)(FRP)}$$

The Core Operating Limits Report contains the TLHGR valves for all resident fuel types.

If during operation, the FDLRC exceeds 1.0 when operating above 25% rated thermal power, either:

a. The APRM scram and rod block settings shall be reduced to the values given by the equations in Specifications 2.1.A.1 and 2.1.B. This may be accomplished by increasing APRM gains as described therein.

b. The power distribution shall be changed such that the FDLRC no longer exceeds 1.0.

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

K. Transient Linear Heat Generation Rate (LHGR)

The fuel design limiting ratio for centerline melt (FDLRC) shall be checked daily during reactor operation at greater than or equal to 25% rated thermal power.

Figures 3.5-1 (Sheets 1 through 3), 3.5-1A and 3.5-1B

Deleted

3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

L. Minimum Critical Power Ratio (MCPR)

1. During steady state operation at all core flows in manual or auto flow control, MCPR shall be greater than or equal to the MCPR Operating Limit obtained using the appropriate flow and scram time dependancy conditions given in the Core Operating Limits Report.

2. During Single Loop operation, the rated flow MCPR operating limit shall be increased by an additive factor of 0.01.

If at any time during steady state power operation, it is determined that the limiting value for MCPR is being exceeded, action

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

L. 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.

Figure 3.5-2 (sheets 1 and 2)

Deleted

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

planar LHGR is sufficient to assure that calculated temperatures are below the 10 CFR 50, Appendix K limit.

The calculational procedure used to establish the maximum average planar LHGR values uses ANF calculational models which are consistent with the requirements of Appendix K 10 CFR 50. The approved calculational models are listed in Specification 6.6.A.4.

ANF has analyzed the effects that Single Loop Operation has on LOCA events. For breaks in the idle loop, the above Dual Loop Operation results are conservative. For breaks in the active loop, the event is more severe primarily due to a more rapid loss of core flow. By applying an appropriate multiplicative reduction factor to the results of the previous analyses, all applicable criteria are met.

J. Local Steady State LHGR

This specification assures that the maximum linear heat generation rate in any fuel rod is less than the design linear heat generation rate even if fuel pellet densification is postulated. This provides assurance that the fuel end-of-life steady state criteria are met.

K. Local Transient LHGR

This specification provides assurance that the fuel will neither experience centerline melt nor exceed 1% plastic cladding strain for transient overpower events beginning at any power and terminating at 120% of rated thermal power.

L. Minimum Critical Power Ratio (MCPR)

The steady-state values for MCPR specified in the Specification were determined using ANF NRC-approved thermal limits methodology described in the ANF NRC-approved methodology listed in Specification 6.6.A.4. 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.

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

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 ANF NRC-approved methodology and appropriate MCPR limits consistent with the ANF NRC-approved 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 or Technical Specification limits 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 as directed by the nuclear fuel vendor in accordance with station procedures.

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 the Core Operating Limits Report. For Automatic Flow Control, in addition to protecting the MCPR Safety Limit during the flow 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 the Core Operating Limits Report.

Analyses have demonstrated that transient events in Single Loop Operation are bounded by those at rated conditions; however, due to the increase in the MCPR fuel cladding integrity safety limit in Single Loop Operation, an equivalent adder must be uniformly applied to the rated flow MCPR LCO to maintain the same margins to the MCPR fuel cladding integrity safety limit.

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

M. 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 (Cont'd.)

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 Steady State LHGR

The FDLRX for all 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. Local Transient LHGR

The fuel design limiting ratio for centerline melt (FDLRC) shall be checked daily during reactor operation at greater than or equal to 25% power to determine if fuel burnup or control rod movement has caused changes in power distribution. The FDLRC limit is designed to protect against centerline melting of the fuel during anticipated operational occurrences.

L. Minimum Critical Power Ratio (MCPR)

At core thermal power levels less than or equal to 25 percent, 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 reduced flow correction applied to the LCO provides margin for flow increase from low flows.

M. Flood Protection

The watertight bulkhead door and the penetration seals for pipes and cables penetrating the vault walls have been designed

3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

e. The suction valve in the idle loop shall be closed and electrically isolated except when the idle loop is being prepared for return to service; and

f. If the tripped pump is out of service for more than 24 hours, implement the following additional restrictions:

i. The flow biased RBM Rod Block LSSS shall be reduced by 4.0% (Specification 3.2.C.1);

ii. The flow biased APRM Rod Block LSSS shall be reduced by 3.5% (Specification 2.1.B);

iii. The flow biased APRM scram LSSS shall be reduced by 3.5% (Specification 2.1.A.1);

iv. The MCPR Safety Limit shall be increased by 0.01 (Specification 1.1.A);

v. The rated flow MCPR Operating Limit shall be increased by 0.01 (Specification 3.5.L.2);

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

vi. The MAPLHGR Operating Limit shall be reduced by the appropriate multiplicative factor from the Core Operating Limits Report (Specification 3.5.I). If, concurrently, one Automatic Pressure Relief Subsystem relief valve is out-of-service, the MAPLHGR Operating Limit shall be reduced by the appropriate multiplicative factor from the Core Operating Limits Report.

4. Core thermal power shall not exceed 25% of rated without forced recirculation. If core thermal power is greater than 25% of rated without forced recirculation, action shall be initiated within 15 minutes to restore operation to within the prescribed limits and core thermal power shall be returned to within the prescribed limit within two (2) hours.

I. Snubbers (Shock Suppressors)

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

I. Snubbers (Shock) Suppressors)

The following surveillance requirements apply to safety related snubbers.

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

In addition, during the start-up of Dresden Unit 2, it was found that a flow mismatch between the two sets of jet pumps caused by a difference in recirculation loops could set up a vibration until a mismatch in speed of 27% occurred. The 10% and 15% speed mismatch restrictions provide additional margin before a pump vibration problem will occur.

Reduced flow MCPR Operating Limits for Automatic Flow Control are not applicable for Single Loop Operation. Therefore, sustained reactor operation under such conditions is not permitted.

Regions I and II of Figure 3.6.2 represent the areas of the power/flow map with the least margin to stable operation. Although calculated decay ratios at the intersection of the natural circulation flow line and the APRM Rod Block line indicate that substantial margin exists to where unstable operation could be expected. Specifications 3.6.H.3.b,, 3.6.H.3.c. and 4.6.H.3. provide additional assurance that if unstable operation should occur, it will be detected and corrected in a timely manner.

During the starting sequence of the inoperable recirculation pump, restricting the operable recirculation pump speed below 65% of rated prevents possible damage to the jet pump riser braces due to excessive vibration.

The closure of the suction valve in the idle loop prevents the loss of LPCI through the idle recirculation pump into the downcomer.

Analyses have been performed which support indefinite operation in single loop provided the restrictions discussed in Specification 3.6.H.3.d. are implemented within 24 hours.

The LSSSs are corrected to account for backflow through the idle jet pumps above 40% of rated recirculation pump speed. This assures that the original drive flow biased rod block and scram trip settings are preserved during Single Loop Operation.

The MCPR safety limit has been increased by 0.01 to account for core flow and TIP reading uncertainties which are used in the statistical analysis of the safety limit. In addition, the rated flow MCPR Operating Limit has also been increased by 0.01 to maintain the same margin to the safety limit as during Dual Loop Operation.

3.6 LIMITING CONDITION FOR OPERATION BASES (Cont'd)

The decrease of the MAPLHGR Operating Limit by the multiplicative factor specified in the Core Operating Limits Report accounts for the more rapid loss of core flow during Single Loop Operation than during Dual Loop Operation.

The more conservative MAPLHGR reduction factors in the Core Operating Limits Report are applied if one relief and one recirculation loop are inoperable at the same time. The small break LOCA is the concern for one relief valve out-of-service; the large break LOCA is the concern for Single Loop Operation. Selecting the more conservative MAPLHGR multipliers will cover both the relief valve out-of-service and Single Loop Operation.

Specification 3.6.H.4 increased the margin of safety for thermal-hydraulic stability and for startup of recirculation pumps from natural circulation conditions.

I. Snubbers (Shock Suppressors)

Snubbers are designed to prevent unrestrained pipe motion under dynamic loads as might occur during an earthquake or severe transient while allowing normal thermal motion during startup and shutdown. The consequence of an inoperable snubber is an increase in the probability of structural damage to piping as a result of a seismic or other event initiating dynamic loads. It is therefore required that all snubbers required to protect the primary coolant system or any other safety system or component be operable during reactor operation.

Because the snubber protection is required only during low probability events, a period of 72 hours is allowed for repairs or replacements. In case a shutdown is required, the allowance of 36 hours to reach a cold shutdown condition will permit an orderly shutdown consistent with standard operating procedures. Since plant startup should not commence with knowingly defective safety related equipment, Specification 3.6.I.4 prohibits startup with inoperable snubbers.

When a snubber is found inoperable, a review shall be performed to determine the snubber mode of failure. Results of the review shall be used to determine if an engineering evaluation of the safety-related system or component is necessary. The engineering evaluation shall determine whether or not the snubber mode of failure has imparted a significant effect or degradation on the support component or system.

All safety related hydraulic snubbers are visually inspected for overall integrity and operability. The inspection will include verification of proper orientation, adequate hydraulic fluid level and proper attachment of snubber to piping and structures.

All safety related mechanical snubbers are visually inspected for overall integrity and operability. The inspection will include verification of proper orientation and attachments to the piping and anchor for indication of damage or impaired operability.

6.0 ADMINISTRATIVE CONTROLS (Cont'd.)

4. Core Operating Limits Report

a. Core operating limits shall be established and documented in the Core Operating Limits Report before each reload cycle or any remaining part of a reload cycle for the following:

1) The Control Rod Withdrawal Block Instrumentation for Table 3.2-3 of Specification 3.2.C.

2) The Average Planar Linear Heat Generation Rate (APLHGR) Limit and associated APLHGR multipliers for Specifications 3.5.I, 3.5.D.2, and 3.6.H.3.f.

3) The Local Steady State Linear Heat Generation Rate (LHGR) for Specification 3.5.J.

4) The Local Transient Linear Heat Generation Rate (LHGR) for Specification 3.5.K.

5) The Minimum Critical Power Operating Limit for Specification 3.5.L. This includes rated and off-rated flow conditions.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC in the latest approved revision or supplement of the topical reports describing the methodology. For Dresden Unit 2, the topical reports are:

1) XN-NF-512(P)(A), "XN-3 Critical Power Correlation.

2) XN-NF-524(P)(A), "Exxon Nuclear Critical Power Methodology for Boiling Water Reactors".

3) XN-NF-79-71(P)(A), "Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors".

4) XN-NF-80-19(P)(A), "Exxon Nuclear Methodology for Boiling Water Reactors".

5) XN-NF-85-67(P)(A), "Generic Mechanical Design for Exxon Nuclear Jet Pump Boiling Water Reactors Reload Fuel".

6) XN-NF-81-22(P)(A), "Generic Statistical Uncertainty Analysis Methodology".

6.0 ADMINISTRATIVE CONTROLS (Cont'd.)

c. The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

d. The Core Operating Limits Report, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

B. Reportable Events

Reportable events will be submitted as required by 10CFR 50.73.

C. Unique Reporting Requirements

1. Radioactive Effluent Release Report (Semi-Annual)

A report shall be submitted to the Commission within 60 days after January 1 and July 1 of each year specifying the quantity of each of the principal radionuclides released to unrestricted areas in liquid and gaseous effluents during the previous 6 months. The format and content of the report shall be in accordance with Regulatory Guide 1.21 (Revision 1) dated June 1974. Any changes to the PCP shall be included in this report.

2. Environmental Radioactivity Data (Annual Report)

a. Standard Radiological Monitoring Program

(1) Non-Routine Report

(a) If a confirmed measured radionuclide concentration in an environmental sampling medium averaged over any calendar quarter sampling period exceeds the reporting level given in Table 4.8-1 and if the radioactivity is attributable to plant operation, a written report shall be submitted to the Regional Administrator of the NRC Regional Office, with a copy of the Director, Office of Nuclear Reactor Regulation, within 30 days from the end of the quarter. When more than one of the radionuclides in Table 4.8-1 are detected in the medium, the reporting level shall have been exceeded if

6.0 ADMINISTRATIVE CONTROLS (Cont'd.)

$\Sigma C_i/(RL)_i$ is equal to or greater than 1

where C is the concentration of the ith radionuclide in the medium and RL is the reporting level of radionuclide i.

(b) If radionuclides other than those in Table 4.8-1 are detected and are due to plant effluents, a reporting level is exceeded if the potential annual dose to an individual is equal to or greater than the design objective doses of 10 CFR 50, Appendix I.

(c) This report shall include an evaluation of any release conditions, environmental factors, or other aspects necessary to explain the anomalous effect.

(2) Annual Operating Report

An annual report containing the data taken in the standard radiological monitoring program (Table 4.8-1) shall be submitted by March 31 of the next year. The content of the report shall include:

(a) Results of environmental sampling summarized on a quarterly basis following the format of Regulatory Guide 4.8 Table 1 (December 1975); (individual sample results will be retained at the station);

In the event that some results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. Summaries, interpretations, and analysis of trends of the results are to be provided.

(b) An assessment of the monitoring results and radiation dose via the principal pathways of exposure resulting from plant emissions of radioactivity including the maximum noble gas gamma and beta air doses in the unrestricted area. The assessment of radiation doses shall be performed in accordance with the ODCM.

6.0 ADMINISTRATIVE CONTROLS (Cont'd.)

(c) Results of the census to determine the locations of animals producing milk for human consumption, and the pasture season feeding practices at dairies in the monitoring program.

(d) The reason for the omission if the nearest dairy to the station is not in the monitoring program. (Table 4.8-5)

(e) An annual summary of meteorological conditions concurrent with the releases of gaseous effluents in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability.

(f) The results of the interlaboratory comparison program described in Section 3.8.E.7.

(g) The results of the 40 CFR 190 uranium fuel cycle dose analysis for each calendar year.

(h) A summary of the monitoring program, including maps showing sampling locations and tables giving distance and direction of sampling locations from the station.

3. Special Reports

Special reports shall be submitted as indicated in Table 6.6.1.

6.7 Environmental Qualification

A. By no later than June 30, 1982 all safety-related electrical equipment in the facility shall be qualified in accordance with the provisions of Division of Operating Reactors "Guidelines for Evaluating Environmental Qualification of Class IE Electrical Equipment in Operating Reactors" (DOR Guidelines); or, NUREG-0588 "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment", December 1979. Copies of these documents are attached to Order for Modification of License DPR-19 dated October 24, 1980.

B. By no later than December 1, 1980, complete and auditable records must be available and maintained at a central location which describe the environmental qualification method used for all safety-related electrical equipment in sufficient detail to document the degree of compliance with the DOR Guidelines or

TABLE 6.6.1

SPECIAL REPORTS

	AREA	SPECIFICATION REFERENCE	SUBMITTAL DATE
a.	Primary Coolant leakage to Drywell (3)	3.6.D Bases	5 years (1)
b.	In-Service Inspection Evaluation (3)	3.6.F Bases	5 years (1)
c.	Evaluation of Economic Generation Control System (EGCS) operation (3)	3.3.G Bases	Upon completion of initial testing
d.	Failed Fuel Detection (3)	3.2 Bases	5 years (1)
e.	Main Steam Line Leakage to Steam Tunnel (3)	3.6.D Bases	5 years (1)
f.	In-service Inspection Development (3)	3.6.F Bases	5 years (1)
g.	In-Service Inspection of Sensitized Stainless Steel Components (2)	4.6.F	4 years (1)
h.	Secondary Containment Leak Rate Test (3)	3.7.C.1	within 90 days after completion of each test
i.	Radioactive Source Leak Testing (4)	3.8.F	Annual Report

NOTES:

1. The report shall be submitted within the period of time,listed based on the commercial service date as the starting point.

2. Dresden 2 only

3. Dresden 2 and 3 only.

4. The report is required only if the tests reveal the presence of 0.005 microcuries or more of removable contamination.

6.0 ADMINISTRATIVE CONTROLS (Cont'd.)

NUREG-0588. Thereafter, such records should be updated and maintained current as equipment is replaced, further tested, or otherwise further qualified.

6.8 Offsite Dose Calculation Manual (ODCM)

A. The ODCM shall describe the methodology and parameters to be used in the calculation of offsite doses due to radioactive gaseous and liquid effluents and in the calculation of gaseous and liquid effluent monitoring instrumentation alarm/trip setpoints consistent with the applicable LCO's contained in these Technical Specifications. Methodologies and calculational procedures acceptable to the Commission are contained in NUREG-0133.

The ODCM shall be submitted to the Commission at the time of proposed Radiological Effluent Technical Specifications and shall be subject to review and approval by the Commission prior to implementation.

B. Licensee initiated changes to the ODCM may be made provided the change:

1. Shall be submitted to the Commission by inclusion in the Monthly Operating Report pursuant to Specification 6.6.A.3. within 90 days of the date the change(s) was made effective and shall contain:

a. Sufficiently detailed information to support the change. Information submitted should consist of a package of those pages of the ODCM to be changed together with appropriate analyses or evaluations justifying the change(s);

b. A determination that the change will not reduce the accuracy or reliability of dose calculations or setpoint determinations; and

c. Documentation of the fact that the change has been reviewed and found acceptable by the On-site Review Function.

2. Shall become effective upon review and acceptance by the On-Site Review Function.

6.0 ADMINISTRATIVE CONTROLS (Cont'd.)

6.9 Process Control Program (PCP)

A. The PCP shall contain the sampling, analysis, and formulation determination by which solidification of radioactive wastes from liquid systems is assured.

B. The PCP shall be approved by the Commission prior to implementation.

C. Licensee initiated changes may be made to the PCP provided the change:

1. Shall be submitted to the Commission in the Radioactive Effluent Release Report for the period in which the change was made and shall contain:

a. Sufficiently detailed information to support the change;

b. A determination that the change did not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes; and

c. Documentation that the change has been reviewed and found acceptable by the On-site Review Function.

2. Shall become effective upon review and acceptance by the On-site Review Function.

6.10 Major Changes to Radioactive Waste Treatment Systems (Liquid, Gaseous, Solid) (See note below)

A. Licensee initiated major changes to the radioactive waste systems may be made provided:

1. The change is reported in the Monthly Operating Report for the period in which the evaluation was reviewed by the On-site Review Function. The discussion of each change shall contain:

a. A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR 50.59;

b. Sufficient detailed information to support the reason for the change;

c. A detailed description of the equipment, components, and process involved and the interfaces with other plant systems;

Note: Licensee may choose to submit this information as part of the annual FSAR update.

6.0 ADMINISTRATIVE CONTROLS (Cont'd.)

d. An evaluation of the change which shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the license application and amendments;

e. A comparison of the predicted releases of radioactive materials in liquid and gaseous effluents and in solid waste to the actual releases for the period in which the changes were made;

f. An estimate of the exposure to plant operating personnel as a result of the change; and

g. Documentation of the fact that the change was reviewed and found acceptable by the On-site Review Function.

2. The change shall become effective upon review and acceptance by the On-site Review Function.

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-249

DRESDEN NUCLEAR POWER STATION, UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 105
License No. DPR-25

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

   A. The application for amendment by the Commonwealth Edison Company (the licensee) dated July 11, 1989, as supplemented by August 14, 1989, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 3.B. of Facility Operating License No. DPR-25 is hereby amended to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 105, 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 to be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

John W. Craig, Director
Project Directorate III-2
Division of Reactor Projects - III,
IV, V and Special Projects
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 8, 1990

ATTACHMENT TO LICENSE AMENDMENT NO. 105

FACILITY OPERATING LICENSE DPR-25

DOCKET NO. 50-249

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change.

REMOVE	INSERT
vi	vi
viii	viii
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1.0-5	1.0-5
B 1/2.1-6	B 1/2.1-6
B 1/2.1-7	B 1/2.1-7
B 1/2.1-8	B 1/2.1-8
B 1/2.1-10	B 1/2.1-10
B 1/2.1-11	B 1.2.1-11
B 1/2.2-4	B 1/2.2-4
3/4.2-12	3/4.2-12
3/4.3-11	3/4.3-11
B 3/4.3-16	B 3/4.3-16
B 3/4.3-17	B 3/4.3-17
B 3/4.3-19	B 3/4.3-19
B 3/4.3-20	B 3/4.3-20
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3/4.5-20	------
3/4.5-21	------
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B 3/4.5-36	B 3/4.5-36
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B 3/4.5-38	B 3/4.5-38
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1.0 DEFINITIONS

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

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

B. Core Operating Limits Report (COLR) - The Core Operating Limits Report is the unit specific document that provides core operating limits for the current operating reload cycle. These cycle specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.6.A.4. Plant operation within these operating limits is addressed in individual specifications.

C. Critical Power Ratio (CPR) - The critical power ratio is the ratio of that assembly power which causes some point in the assembly to experience transition boiling to the assembly power at the reactor condition of interest as calculated by application of the ANF NRC-approved correlation.

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

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

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

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

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

I. Limiting Conditions for Operation (LCO) - The limiting conditions for operation specify the minimum acceptable levels of system performance necessary to assure safe startup and operation of the facility. When these conditions are met, the

1.0 DEFINITIONS (Cont'd.)

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

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

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

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

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

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

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

DD. 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. Fuel Design Limiting Ratio (FDLRX) - The fuel design limiting ratio is the limit used to assure that the fuel operates within the end-of-life steady state design criteria. FDLRX assures acceptable end-of-life conditions by, among other items, limiting the release of fission gas to the cladding plenum.

GG. Dose Equivalent I-131 - That concentration of I-131 (microcurie/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

1.1 SAFETY LIMIT BASES

FUEL CLADDING INTEGRITY

The fuel cladding integrity limit is set such that no calculated fuel damages 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 greater than the MCPR fuel cladding integrity safety limit represents a conservative margin relative to the conditions required to maintain fuel cladding integrity by assuring that the fuel does not experience transition boiling.

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 corrosions 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 perforation signals 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. The MCPR fuel cladding integrity Safety Limit assures that during normal operation and during anticipated operational occurrences, at least 99.9% of the fuel rods in the core do not experience transition boiling.

A. Reactor Pressure greater than 800 psig and Core Flow greater than 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

1.1 SAFETY LIMIT BASES (Cont'd.)

power ratio (CPR) which is the ratio of the bundle power which would produce the onset of transition boiling divided by the actual bundle power. The minimum value of this ratio for any bundle in the core is the Minimum Critical Power Ratio (MCPR). It is assumed that the plant operation is controlled to the nominal protective setpoints via the instrumented variables. (Figure 2.1-3).

The MCPR Fuel Cladding Integrity Safety Limit assures sufficient conservatism in the operating MCPR limit that in the event of an anticipated operational occurrence from the limiting condition for operation, at least 99.9% of the fuel rods in the core would be expected to avoid boiling transition. The margin between calculated boiling transition (MCPR=1.00) and the MCPR Fuel Cladding Integrity Safety Limit is based on a detailed statistical procedure which considers the uncertainties in monitoring the core operating state. One specific uncertainty included in the safety limit is the uncertainty inherent in the ANF NRC-approved critical power correlation. Refer to Specification 6.6.A.4 for the methodology used in determining the MCPR Fuel Cladding Integrity Safety Limit.

The ANF NRC-approved critical power correlation is based on a significant body of practical test data, providing a high degree of assurance that the critical power as evaluated by the correlation is within a small percentage of the actual critical power being estimated. The assumed reactor conditions used in defining the safety limit introduce conservatism into the limit because boundingly high radial power peaking factors and boundingly flat local peaking distributions are used to estimate the number of rods in boiling transition. Still further conservatism is induced by the tendency of the ANF NRC-approved correlation to overpredict the number of rods in boiling transition. These conservatisms and the inherent accuracy of the ANF NRC-approved correlation provide a reasonable degree of assurance that during sustained operation at the MCPR Fuel Cladding Integrity Safety Limit there would be no transition boiling in the core. If boiling transition were to occur, however, there is reason to believe that the integrity of the fuel would not necessarily be compromised. Significant test data accumulated by the U.S. Nuclear Regulatory Commission and private organizations indicate that the use of a boiling transition limitation to protect against cladding failure is a very conservative approach; much of the data indicates that LWR fuel can survive for an extended period in an environment of transition boiling.

During Single Loop Operation, the MCPR safety limit is increased by 0.01 to conservatively account for increased uncertainties in the core flow and TIP measurements.

1.1 SAFETY LIMIT BASES (Cont'd.)

If the reactor pressure should ever exceed the limit of applicability of the ANF NRC-approved critical power correlation as defined in the ANF NRC-approved methodology listed in Specification 6.6.A.4, it would be assumed that the MCPR Fuel Cladding Integrity Safety Limit had been violated. This applicability pressure limit is higher than the pressure safety limit specified in Specification 1.2.

Fuel design criteria have been established to provide protection against fuel centerline melting and 1% plastic cladding strain during transient overpower conditions throughout the life of the fuel. To demonstrate compliance with these criteria, fuel rod centerline temperatures are determined at 120% overpower conditions as a check against calculated centerline melt temperatures. FDLRC is incorporated to protect the above criteria at all power levels considering events which will cause the reactor power to increase to 120% of rated thermal power.

B. Core Thermal Power Limit (Reactor Pressure less than 800 psia)

At pressures below 800 psia, the core elevation pressure drop (0 power, 0 flow) is greater than 4.56 psi. At low powers and flows this pressure differential is maintained in the bypass region of the core. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low powers and flows will always be greater than 4.56 psi. Analyses show that with a flow of 28×10^3 lbs/hr bundle flow, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow

1.1 SAFETY LIMIT BASES (Cont'd.)

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. See the ANF NRC-approved methodology listed in Specification 6.6.A.4.

Conservatism is incorporated into the transient analyses which define the MCPR operating limits. Variables which inherently possess little or no uncertainty or whose uncertainty has little or no effect on the outcome of the limiting transient are selected at bounding values. Variables which possess significant uncertainty that may have undesirable effects on thermal margins are addressed statistically. Statistical methods used in the transient analyses are described in the ANF NRC-approved methodology listed in Specification 6.6.A.4. The MCPR operating limits are established such that the occurrence of the limiting transient will not result in the violation of the MCPR Fuel Cladding Integrity Safety Limit in at least 95% of the random statistical combinations of uncertainties. In general, the variables with the greatest statistical significance to the consequences of anticipated operational occurrences are the reactivity feedback associated with the formation and removal of coolant voids and the timing of the control rod scram.

2.1 LIMITING SAFETY SYSTEM SETTING BASES (Cont'd.)

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.L 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 during dual loop operation or 116.5 percent during single loop operation, 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.

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.

2.2 LIMITING SAFETY SYSTEM SETTING BASES

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 over-pressurizing the reactor pressure vessel and thus exceeding the pressure safety limit. The pressure scram is available as a backup protection to the direct valve position trip scrams and 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 ANF NRC-approved methodology listed in Specification 6.6.A.4 for each reload to assure that the pressure safety limit is not exceeded.

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)	
	Dual Loop Operation	Less than or equal to (.58 W_D plus 50)/FDLRC (See Note 2)
	Single Loop Operation	Less than or equal to (.58 W_D plus 46.5)/FDLRC (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)	
	Dual Loop Operation	See the Core Operating Limits Report
	Single Loop Operation	See the Core Operating Limits Report
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	(See Note 4)
2 (5) (6)	SRM upscale	Less than or equal to 10^5 counts/sec.
1	Scram discharge volume water level - high	Less than or equal to 25 gallons

Notes: (See Next Page)

3.3 LIMITING CONDITION FOR OPERATION (Cont'd.)

2. The maximum scram insertion time for 90% insertion of any operable control rod shall not exceed 7.00 seconds.

D. Control Rod Accumulators

At all reactor operating pressures, a rod accumulator may be inoperable provided that no other control rod in the nine-rod square array around this rod has a:

1. Inoperable accumulator,
2. Directional control valve electrically disarmed while in a non-fully inserted position.

4.3 SURVEILLANCE REQUIREMENT (Cont'd.)

2. At 16 week intervals, at least 50% of the control rod drives shall be tested as in 4.3.C.1 so that every 32 weeks all of the control rods shall have been tested. Whenever 50% or more of the control rod drives have been tested, an evaluation shall be made to provide reasonable assurance that proper control rod drive performance is being maintained.

3. Following completion of each set of scram testing as described above, the results will be compared against the average scram speed distribution used in the transient analysis to verify the applicability of the current MCPR Operating Limit. Refer to Specification 3.5.L.

D. Control Rod Accumulators

Once a shift check the status of the pressure and level alarms for each accumulator.

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

3. The operability of the scram discharge volume vent and drain valves assures the proper venting and draining of the volume. This ensures that water accumulation does not occur which would cause an early termination of control rod movement during a full core scram. These specifications provide for the periodic verification that the valves are open and for testing of these valves under reactor scram conditions during each Refueling Outage.

B. Control Rod Withdrawal

1. Control rod dropout accidents as discussed in the ANF NRC-approved methodology, listed in Specification 6.6.A.4, 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.

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

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

withdrawn could not be worth enough to cause the rod drop accident design limit of 280 cal/gm to be exceeded if they were to drop out of the core in the manner defined for the Rod Drop Accident. 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 RWM 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, 425 cal/gm, at which rapid fuel dispersal and primary system damage have been found to occur based on experimental data as is discussed in the ANF NRC-approved methodology listed in Specification 6.6.A.4.

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.

Parametric Control Rod Drop Accident analyses have shown that for wide ranges of key reactor parameters (which envelope the operating ranges of these variables), the fuel enthalpy rise during a postulated control rod drop accident remains considerably lower than the 280 cal/gm limit. For each operating cycle, cycle-specific parameters such as maximum control rod worth, Doppler coefficient effective delayed neutron fraction and maximum four-bundle local peaking factor are compared with the results of the parametric analyses to determine the peak fuel rod enthalpy rise. This value is then compared against the Technical Specification limit of 280 cal/gm to demonstrate compliance for each operating cycle. If cycle specific values of the above parameters are outside the range assumed in the parametric analyses, an extension of the analysis or a cycle specific analysis may be required. Conservatism present in the analysis, results of the parametric studies, and a detailed description of the methodology for performing the Control Rod Drop Accident analysis are provided in the ANF NRC-approved methodology listed in Specification 6.6.A.4.

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

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 pattern, 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 performance of the control rod insertion system is analyzed to verify the system's ability to bring the reactor subcritical at a rate fast enough to prevent violation of the MCPR Fuel Cladding Integrity Safety Limit and thereby avoid fuel damage. The analyses demonstrate that if the reactor is operated within the limitations set in Specification 3.5.L the negative reactivity insertion rates associated with the observed scram performance (as adjusted for statistical variation in the observed data) result in protection of the MCPR safety limit.

In the analytical treatment of most 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-energizes 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

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

analyses, and is also included in the allowable scram insertion times specified in Specification 3.3.C. In the statistical treatment of the limiting transients, a statistical distribution of total scram delay is used rather than the bounding value described above.

The performance of the individual control rod drives is monitored to assure that scram performance is not degraded. Fifty percent of the control rod drives in the reactor are tested every sixteen weeks to verify adequate performance. Observed plant data or Technical Specification limits (Specification 3.3.C) were used to determine the average scram performance used in the transient analyses and the results of each set of control rod scram tests performed per Specification 3.3.C during the current cycle are compared against earlier results to verify that the performance of the control rod insertion system has not changed significantly. If a test performed per Specification 3.3.C should be determined to fall outside of the statistical population defining the scram performance characteristics used in the transient analyses, a re-determination of thermal margin requirements is undertaken as required by Specification 3.5.L. A smaller test sample than that required by Specification 3.3.C is not statistically significant and should not be used in the re-determination of thermal margins. Control rod drives with excessive scram times can be fully inserted into the core and deenergized in the manner of an inoperable rod drive provided the allowable number of inoperable control rod drives is not exceeded. In this case, the scram speed of the drive shall not be used as a basis in the re-determination of thermal margin requirements.

The scram times for all control rods are measured at the time of each refueling outage. Experience with the plant has shown that control drive insertion times vary little through the operating cycle; hence no reassessment of thermal margin requirements is expected under normal conditions. 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 second is greater than 0.999 for a normal distribution.

3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

operation is permissible only during the succeeding seven (7) days provided that during such time the HPCI subsystem is operable. If the appropriate MAPLHGR reduction factors (multipliers) are applied to the MAPLHGR Limits, the Automatic Pressure Relief Subsystem of ECCS shall be considered operable. The MAPLHGR Limits and the appropriate MAPLHGR reductions factors are found in the Core Operating Limits Report.

3. From and after the date that two relief valves are found or made to be inoperable, reactor operation is permissible only during the succeeding seven days provided that during such time the HPCI subsystem is operable and the multipliers specified in 3.5.D.2 are applied.

4. If the requirements of 3.5.D.1 cannot be met, an orderly shutdown shall be initiated and the reactor pressure shall be reduced to below 150 psig within 24 hours.

E. Isolation Condenser System

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

E. Surveillance of the Isolation Condenser System shall be performed as follows:

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 shall not exceed the MAPLHGR Limits. For Single Loop Operation (SLO), the MAPLHGR Limits shall be decreased by the SLO MAPLHGR multiplicative factor(s). If concurrent with SLO, one Automatic Pressure Relief Subsystem Relief Valve is Out of Service (RVOOS), the MAPLHGR Limits shall be decreased by the RVOOS MAPLHGR Multiplicative factor(s). The MAPLHGR Limits and multiplicative factors are specified in the Core Operating Limits Report. 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.

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

I. Average Planar Linear Heat Generation Rate (APLHGR)

The APLHGR for each type of fuel shall be determined daily during reactor operation at greater than or equal to 25% rated thermal power.

3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

J. LOCAL STEADY STATE LHGR

During steady state power operation above 25% rated thermal power, the linear heat generation rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed its maximum steady state LHGR (SLHGR) value shown in the Core Operating Limits Report. That is, the Fuel Design Limiting Ratio (FDLRX) shall not be greater than 1.0 where

$$FDLRX = \frac{LHGR}{SLHGR}$$

If at any time during operation above 25% rated thermal power, it is determined by normal surveillance that FDLRX for any fuel assembly exceeds 1.0, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the FDLRX 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.)

J. Linear Heat Generation Rate (LHGR)

The Fuel Design Limiting Ratio (FDLRX) shall be checked daily during reactor operation at greater than or equal to 25% rated thermal power.

3.5 LIMITING CONDITION FOR OPERATION (Cont'd)

K. Local Transient LHGR

At any time during power operation, above 25% rated thermal power the fuel design limiting ratio for centerline melt (FDLRC) shall not be greater than 1.0, where

$$FDLRC = \frac{(LHGR)\ (1.2)}{(TLHGR)\ (FRP)}$$

The Core Operating Limits Report contains the TLHGR valves for all resident fuel types.

If during operation, the FDLRC exceeds 1.0 when operating above 25% rated thermal power, either:

a. The APRM scram and rod block settings shall be reduced to the values given by the equations in Specifications 2.1.A.1 and 2.1.B. This may be accomplished by increasing APRM gains as described therein.

b. The power distribution shall be changed such that the FDLRC no longer exceeds 1.0.

4.5 SURVEILLANCE REQUIREMENT (Cont'd)

K. Transient Linear Heat Generation Rate (LHGR)

The fuel design limiting ratio for centerline melt (FDLRC) shall be checked daily during reactor operation at greater than or equal to 25% rated thermal power

Figure 3.5.1 (sheets 1 and 2), Figure 3.5-1A, Figure 3.5-1B

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3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

L. Minimum Critical Power Ratio (MCPR)

1. During steady state operation at all core flows in manual or auto flow control. MCPR shall be greater than or equal to the MCPR Operating Limit obtained using the appropriate flow and scram time dependancy conditions given in the Core Operating Limits Report.

2. During Single Loop Operation, the rated flow MCPR operating limit shall be increased by an additive factor of 0.01.

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

L. 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.

Figure 3.5-2 (Sheets 1 through 3)

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

planar LHGR is sufficient to assure that calculated temperatures are below the 10 CFR 50, Appendix K limit.

The calculational procedure used to establish the maximum average planar LHGR values uses ANF calculational models which are consistent with the requirements of Appendix K 10 CFR 50. The approved calculational models are listed in Specification 6.6.A.4.

ANF has analyzed the effects Single Loop Operation has on LOCA events. For breaks in the idle loop, the above Dual Loop Operation results are conservative. For breaks in the active loop, the event is more severe primarily due to a more rapid loss of core flow. By applying an appropriate multiplicative reduction factor to the results of the previous analyses, all applicable criteria are met.

J. Local Steady State LHGR

This specification assures that the maximum linear heat generation rate in any fuel rod is less than the design linear heat generation rate even if fuel pellet densification is postulated. This provides assurance that the fuel end-of-life steady state criteria are met.

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

K. Local Transient LHGR

This specification provides assurance that the fuel will neither experience centerline melt nor exceed 1% plastic cladding strain for transient overpower events beginning at any power and terminating at 120% of rated thermal power.

L. Minimum Critical Power Ratio (MCPR)

The steady-state values for MCPR specified in the Specification were determined using ANF NRC-approved thermal limits methodology described in the ANF NRC-approved methodology listed in Specification 6.6.A.4. 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 ANF NRC-approved methodology and appropriate MCPR limits consistent with the the ANF NRC-approved 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 or Technical Specification Limits (Specification 3.3.C) 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 as directed by the nuclear fuel vendor in accordance with station procedures.

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

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 the Core Operating Limits Report. For Automatic Flow Control, in addition to protecting the MCPR Safety Limit during the flow 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 the Core Operating Limits Report.

Analyses have demonstrated that transient events in Single Loop Operation are bounded by those at rated conditions; however, due to the increase in the MCPR fuel cladding integrity safety limit in Single Loop Operation, an equivalent adder must be uniformly applied to the rated flow MCPR LCO to maintain the same margins to the MCPR fuel cladding integrity safety limit.

M. 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 (Cont'd.)

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

J. Local Steady State LHGR

The FDLRX for all fuel shall be checked daily during reactor operation at greater than or equal to 25 percent 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. Local Transient LHGR

The fuel design limiting ratio for centerline melt (FDLRC) shall be checked daily during reactor operation at greater than or equal to 25% power to determine if fuel burnup or control rod movement has caused changes in power distribution. The FDLRC limit is designed to protect against centerline melting of the fuel during anticipated operational occurrences.

L. Minimum Critical Power Ratio (MCPR)

At core thermal power levels less than or equal to 25 percent, 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 reduced flow correction applied to the LCO provides margin for flow increase from low flows.

M. Flood Protection

The watertight bulkhead door and the penetration seals for pipes and cables penetrating the vault walls have been designed

3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

e. The suction valve in the idle loop shall be closed and electrically isolated except when the idle loop is being prepared for return to service; and

f. If the tripped pump is out of service for more than 24 hours, implement the following additional restrictions:

i. The flow biased RBM Rod Block LSSS shall be reduced by 4.0% (Specification 3.2.C.1);

ii. The flow biased APRM Rod Block LSSS shall be reduced by 3.5% (Specification 2.1.B);

iii. The flow biased APRM scram LSSS shall be reduced by 3.5% (Specification 2.1.A.1);

iv. The MCPR Safety Limit shall be increased by 0.01 (Specification 1.1.A);

v. The rated flow MCPR Operating Limit shall be increased by 0.01 (Specification 3.5.L.2);

3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

vi. The MAPLHGR Operating Limit shall be reduced by the appropriate multiplicative factor from the Core Operating Limits Report (Specification 3.5.I). If concurrently, one Automatic Pressure Relief Subsystem relief valve is out-of-service, the MAPLHGR Operating Limit shall be reduced by the appropriate multiplicative factor from the Core Operating Limits Report.

4. Core thermal power shall not exceed 25% of rated without forced recirculation. If core thermal power is greater than 25% of rated without forced recirculation, action shall be initiated within 15 minutes to restore operation to within the prescribed limits and core thermal power shall be returned to within the prescribed limit within two (2) hours.

I. Snubbers (Shock Suppressors)

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

I. Snubbers (Shock Suppressors)

The following surveillance requirements apply to safety related snubbers.

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

In addition, during the start-up of Dresden Unit 2, it was found that a flow mismatch between the two sets of jet pumps caused by a difference in recirculation loops could set up a vibration until a mismatch in speed of 27% occurred. The 10% and 15% speed mismatch restrictions provide additional margin before a pump vibration problem will occur.

Reduced flow MCPR Operating Limits for Automatic Flow Control are not applicable for Single Loop Operation. Therefore, sustained reactor operation under such conditions is not permitted.

Regions I and II of Figure 3.6.2 represent the areas of the power/flow map with the least margin to stable operation. Although calculated decay ratios at the intersection of the natural circulation flow line and the APRM Rod block line indicate that substantial margin exists to where unstable operation could be expected. Specifications 3.6.H.3.b., 3.6.H.3.c. and 4.6.H.3. provide additional assurance that if unstable operation should occur, it will be detected and corrected in a timely manner.

During the starting sequence of the inoperable recirculation pump, restricting the operable recirculation pump speed below 65% of rated prevents possible damage to the jet pump riser braces due to excessive vibration.

The closure of the suction valve in the idle loop prevents the loss of LPCI through the idle recirculation pump into the downcomer.

Analyses have been performed which support indefinite operation in single loop provided the restrictions discussed in Specification 3.6.H.3.d. are implemented within 24 hours.

The LSSSs are corrected to account for backflow through the idle jet pumps above 20-40% of rated recirculation pump speed. This assures that the original drive flow biased rod block and scram trip settings are preserved during Single Loop Operation.

The MCPR safety limit has been increased by 0.01 to account for core flow and TIP reading uncertainties which are used in the statistical analysis of the safety limit. In addition, the rated flow MCPR Operating Limit has also been increased by 0.01 to maintain the same margin to the safety limit as during Dual Loop Operation.

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

The decrease of the MAPLHGR Operating Limit by the multiplicative factor specified in the Core Operating Limits Report accounts for the more rapid loss of core flow during Single Loop Operation than during Dual Loop Operation.

The more conservative MAPLHGR reduction factors in the Core Operating Limits Report are applied if one relief valve and one recirculation loop are inoperable at the same time. The small break LOCA is the concern for one relief valve out-of-service; the large break LOCA is the concern for Single Loop Operation. Selecting the more conservative MAPLHGR multipliers will cover both the relief value out-of-service and Single Loop Operation.

Specification 3.6.H.4 increased the margin of safety for thermal-hydraulic stability and for startup of recirculation pumps from natural circulation conditions.

I. Snubbers (Shock Suppressors)

Snubbers are designed to prevent unrestrained pipe motion under dynamic loads as might occur during an earthquake or severe transient while allowing normal thermal motion during startup and shutdown. The consequence of an inoperable snubber is an increase in the probability of structural damage to piping as a result of a seismic or other event initiating dynamic loads. It is therefore required that all snubbers required to protect the primary coolant system or any other safety system or component be operable during reactor operation.

Because the snubber protection is required only during low probability events, a period of 72 hours is allowed for repairs or replacements. In case a shutdown is required, the allowance of 36 hours to reach a cold shutdown condition will permit an orderly shutdown consistent with standard operating procedures. Since plant startup should not commence with knowingly defective safety related equipment, Specification 3.6.I.4 prohibits startup with inoperable snubbers.

When a snubber is found inoperable, a review shall be performed to determine the snubber mode of failure. Results of the review shall be used to determine if an engineering evaluation of the safety-related system or component is necessary. The engineering evaluation shall determine whether or not the snubber mode of failure has imparted a significant effect or degradation on the support component or system.

All safety related hydraulic snubbers are visually inspected for overall integrity and operability. The inspection will include verification of proper orientation, adequate hydraulic fluid level and proper attachment of snubber to piping and structures.

6.0 ADMINISTRATIVE CONTROLS (Cont'd.)

additional specific details required in license conditions based on other commitments shall be included in this report.

Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial power operation), supplementary reports shall be submitted at least every three months until all three events have been completed.

2. A tabulation shall be submitted on an annual basis of the number of station, utility and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man rem exposure according to work and job functions, (See note); e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignment to various duty functions may be estimates based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific major work functions.

3. Monthly Operating Report

Routine reports of operating statistics and shutdown experiences shall be submitted on a monthly basis to the United States Nuclear Regulatory Commission, Washington, DC 20555, with a copy to the appropriate Regional Administrator, to arrive no later than the 15th of each month following the calendar month covered by the report.

4. Core Operating Report

a. Core operating limits shall be established and documented in the Core Operating Limits Report before each reload cycle or any remaining part of a reload cycle for the following:

1) The Control Rod Withdrawal Block Instrumentation for Table 3.2-3 of Specification 3.2.C.

2) The Average Planar Linear Heat Generation Rate (APLHGR) Limit and associated APLHGR multipliers for Specifications 3.5.I, 3.5.D.2, and 3.6.H.3.f.

Note: This tabulation supplements the requirements of 20.407 of 10 CFR Part 20.

6.0 ADMINISTRATIVE CONTROLS (Cont'd.)

3) The Local Steady State Linear Heat Generation Rate (LHGR) for Specification 3.5.J.

4) The Local Transient Linear Heat Generation Rate (LHGR) for Specification 3.5.K.

5) The Minimum Critical Power Operating Limit for Specification 3.5.L. This includes rated and off-rated flow conditions.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC in the latest approved revision or supplement of the topical reports describing the methodology. For Dresden Unit 3, the topical reports are:

1) XN-NF-512(P)(A), "XN-3 Critical Power Correlation.

2) XN-NF-524(P)(A), "Exxon Nuclear Critical Power Methodology for Boiling Water Reactors".

3) XN-NF-79-71(P)(A), "Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors".

4) XN-NF-80-19(P)(A), "Exxon Nuclear Methodology for Boiling Water Reactors".

5) XN-NF-85-67(P)(A), "Generic Mecahnical Design for Exxon Nuclear Jet Pump Boiling Water Reactors Reload Fuel".

6) XN-NF-81-22(P)(A), "Generic Statistical Uncertainty Analysis Methodology".

c. The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

d. The Core Operating Limits Report, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

B. Reportable Events

Reportable events will be submitted as required by 10 CFR 50.73.

6.0 ADMINISTRATIVE CONTROLS (Cont'd.)

C. Unique Reporting Requirements

1. Radioactive Effluent Release Report (Semi-Annual)

A report shall be submitted to the Commission within 60 days after January 1 and July 1 of each year specifying the quantity of each of the principal radionuclides released to unrestricted areas in liquid and gaseous effluents during the previous 6 months. The format and content of the report shall be in accordance with Regulatory Guide 1.21 (Revision 1) dated June 1974. Any changes to the PCP shall be included in this report.

a. Standard Radiological Monitoring Program

(1) Non-Routine Report

(a) If a confirmed measured radionuclide concentration in an environmental sampling medium averaged over any calendar quarter sampling period exceeds the reporting level given in Table 4.8-1 and if the radioactivity is attributable to plant operation, a written report shall be submitted to the Regional Administrator of the NRC Regional Office, with a copy to the Director, Office of Nuclear Reactor Regulation, within 30 days from the end of the quarter. When more than one of the radionuclides in Table 4.8-1 are detected in the medium, the reporting level shall have been exceeded if

$\Sigma C_i/(RL)_i$ is equal to or greater than 1 where C is the concentration of the ith radionuclide in the medium and RL is the reporting level of radionuclide i.

(b) If radionuclides other than those in Table 4.8-1 are detected and are due to plant effluents, a reporting level is exceeded if the potential annual dose to an individual is equal to or greater than the design objective doses of 10 CFR 50, Appendix I.

2. Environmental Radioactivity Data (Annual Report)

(c) This report shall include an evaluation of any release conditions, environmental factors, or other aspects necessary to explain the anomalous effect.

(2) Annual Operating Report

An annual report containing the data taken in the standard radiological monitoring program (Table 4.8-1) shall be submitted by March 31 of the next year. The content of the report shall include:

6.0 ADMINISTRATIVE CONTROLS (Cont'd.)

(a) Results of environmental sampling summarized on a quarterly basis following the format of Regulatory Guide 4.8 Table 1 (December 1975); (individual sample results will be retained at the station);

In the event that some results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. Summaries, interpretations, and analysis of trends of the results are to be provided.

(b) An assessment of the monitoring results and radiation dose via the principal pathways of exposure resulting from plant emissions of radioactivity including the maximum noble gas gamma and beta air doses in the unrestricted area. The assessment of radiation doses shall be performed in accordance with the ODCM.

(c) Results of the census to determine the locations of animals producing milk for human consumption, and the pasture season feeding practices at dairies in the monitoring program.

(d) The reason for the omission if the nearest dairy to the station is not in the monitoring program. (Table 4.8-5)

(e) An annual summary of meteorological conditions concurrent with the releases of gaseous effluents in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability.

(f) The results of the interlaboratory comparison program described in Section 3.8.E.7.

(g) The results of the 40 CFR 190 uranium fuel cycle dose analysis for each calendar year.

(h) A summary of the monitoring program, including maps showing sampling locations and tables giving distance and direction of sampling locations from the station.

3. Special Reports

Special reports shall be submitted as indicated in Table 6.6.1.

6.7 Environmental Qualification

A. By no later than June 30, 1982 all safety-related electrical equipment in the facility shall be qualified in accordance with the provisions

TABLE 6.6.1

SPECIAL REPORTS

	AREA	SPECIFICATION REFERENCE	SUBMITTAL DATE
a.	Primary Coolant leakage to Drywell (3)	3.6.D Bases	5 years (1)
b.	In-Service Inspection Evaluation (3)	3.6.F Bases	5 years (1)
c.	Evaluation of Economic Generation Control System (EGCS) operation (3)	3.3.G Bases	Upon completion of initial testing
d.	Failed Fuel Detection (3)	3.2 Bases	5 years (1)
e.	Main Steam Line Leakage to Steam Tunnel (3)	3.6.D Bases	5 years (1)
f.	In-service Inspection Development (3)	3.6.F Bases	5 years (1)
g.	In-Service Inspection of Sensitized Stainless Steel Components (2)	4.6.F	4 years (1)
h.	Secondary Containment Leak Rate Test (3)	3.7.C.1	within 90 days after completion of each test
i.	Radioactive Source Leak Testing (4)	3.8.F	Annual Report

NOTES:
1. The report shall be submitted within the period of time listed based on the commercial service date as the starting point.
2. Dresden 2 only
3. Dresden 2 and 3 only.
4. The report is required only if the tests reveal the presence of 0.005 microcuries or more of removable contamination.

6.0 ADMINISTRATIVE CONTROLS (Cont'd.)

of Division of Operating Reactors "Guidelines for Evaluating Environmental Qualification of Class IE Electrical Equipment in Operating Reactors" (DOR Guidelines); or NUREG-0588 "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment", December 1979. Copies of these documents are attached to Order for Modification of License DPR-25 dated October 24, 1980.

B. By no later than December 1, 1980, complete and auditable records must be available and maintained at a central location which describe the environmental qualification method used for all safety-related electrical equipment in sufficient detail to document the degree of compliance with the DOR Guidelines or NUREG-0588. Thereafter, such records should be updated and maintained current as equipment is replaced, further tested, or otherwise further qualified.

6.8 Offsite Dose Calculation Manual (ODCM)

A. The ODCM shall describe the methodology and parameters to be used in the calculation of offsite doses due to radioactive gaseous and liquid effluents and in the calculation of gaseous and liquid effluents monitoring instrumentation alarm/trip setpoints consistent with the applicable LCO's contained in these Technical Specifications. Methodologies and calculational procedures acceptable to the Commission are contained in NUREG-0133.

The ODCM shall be submitted to the Commission at the time of proposed Radiological Effluent Technical Specifications and shall be subject to review and approval by the Commission prior to implementation.

B. Licensee initiated changes to the ODCM may be made provided the change:

1. Shall be submitted to the Commission by inclusion in the Monthly Operating Report pursuant to Specification 6.6.A.3. within 90 days of the date the change(s) was made effective and shall contain:

a. Sufficiently detailed information to support the change. Information submitted should consist of a package of those pages of the ODCM to be changed together with appropriate analyses or evaluations justifying the change(s);

b. A determination that the change will not reduce the accuracy or reliability of dose calculations or setpoint determinations; and

c. Documentation of the fact that the change has been reviewed and found acceptable by the On-site Review Function.

2. Shall become effective upon review and acceptance by the On-site Review Function.

6.0 ADMINISTRATIVE CONTROLS (Cont'd.)

6.9 Process Control Program (PCP)

A. The PCP shall contain the sampling, analysis, and formulation determination by which solidification of radioactive wastes from liquid systems is assured.

B. The PCP shall be approved by the Commission prior to implementation.

C. Licensee initiated changes may be made to the PCP provided the change:

1. Shall be submitted to the Commission in the Radioactive Effluent Release Report for the period in which the change was made and shall contain:

a. Sufficiently detailed information to support the change;

b. A determination that the change did not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes; and

c. Documentation that the change has been reviewed and found acceptable by the On-site Review Function.

2. Shall become effective upon review and acceptance by the On-site Review Function.

6.10 Major Changes to Radioactive Waste Treatment Systems (Liquid, Gaseous, Solid) (see note below)

A. Licensee initiated major changes to the radioactive waste systems may be made provided:

1. The change is reported in the Monthly Operating Report for the period in which the evaluation was reviewed by the On-site Review Function. The discussion of each change shall contain:

a. A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR 50.59;

b. Sufficient detailed information to support the reason for the change;

c. A detailed description of the equipment, components, and process involved and the interfaces with other plant systems;

Note: Licensee may choose to submit this information as part of the annual FSAR update.

6.0 ADMINISTRATIVE CONTROLS (Cont'd.)

d. An evaluation of the change which shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the license application and amendments;

e. A comparison of the predicted releases of radioactive materials in liquid and gaseous effluents and in solid waste to the actual releases for the period in which the changes were made;

f. An estimate of the exposure to plant operating personnel as a result of the change; and

g. Documentation of the fact that the change was reviewed and found acceptable by the On-site Review Function.

2. The change shall become effective upon review and acceptance by the On-site Review Function.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 110 TO PROVISIONAL OPERATING LICENSE NO. DPR-19

AND AMENDMENT NO. 105 TO FACILITY OPERATING LICENSE NO. DPR-25

COMMONWEALTH EDISON COMPANY

DRESDEN NUCLEAR POWER STATION, UNIT NOS. 2 AND 3

DOCKET NOS. 50-237 AND 50-249

1.0 INTRODUCTION

By letter dated October 30, 1989 (Ref. 1), as supplemented by letter dated November 20, 1989 (Ref. 2), Commonwealth Edison Company (the licensee) proposed changes to the Technical Specifications (TS) for the Dresden Nuclear Power Station, Units 2 and 3. The proposed changes would modify specifications having cycle-specific parameter limits by replacing the values of those limits with a reference to the Core Operating Limits Report (COLR) for the values of those limits. The proposed changes also include the addition of the COLR to the Definitions section and to the reporting requirements of the Administrative Controls section of TS. Guidance on the proposed changes was developed by NRC on the basis of the review of a lead-plant proposal submitted on the Oconee plant docket by Duke Power Company. This guidance was provided to all power reactor licensees and applicants by Generic Letter 88-16, dated October 4, 1988 (Ref. 3).

The November 20, 1989 submittal incorporated the referenced topical reports as an integral part of Section 6 to the Technical Specifications. Since this submittal only provided clarification to the October 30, 1989 submittal, it did not change the action as previously noticed, or alter the initial no significant determination published, in the Federal Register on November 29, 1989.

2.0 EVALUATION

The licensee's proposed changes to the TS are in accordance with the guidance provided by Generic Letter 88-16 and are addressed below.

(1) The Definition section of the TS was modified to include a definition of the Core Operating Limits Report that requires cycle/reload-specific parameter limits to be established on a unit-specific basis in accordance with an NRC approved methodology that maintains the limits of the safety analysis. The definition notes that plant operation within these limits is addressed by individual specifications.

(2) The following specifications were revised for both Dresden 2 and Dresden 3 to replace the values of cycle-specific parameter limits with a reference to the COLR that provides these limits.

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(a) Specification 3.2.C (Table 3.2-3)

The Rod Block Monitor (RBM) upscale setpoint relationships for this specification are specified in the COLR.

(b) Specification 3.5.I, 3.5.D.2 and 3.6.H.3.f

The Average Planar Linear Heat Generation Rate (APLHGR) limits and associated APLHGR multipliers for these specifications are specified in the COLR.

(c) Specification 3.5.J

The Local Steady State Linear Heat Generation Rate (LHGR) limit for this specification is specified in the COLR.

(d) Specification 3.5.K

The Local Transient Linear Heat Generation Rate (LHGR) limit for this specification is specified in the COLR.

(e) Specification 3.5.L

The Minimum Critical Power Ratio (MCPR) operating limits for this specification are specified in the COLR.

The bases of affected specifications have been modified by the licensee to include appropriate reference to the COLR. Based on our review, we conclude that the changes to these bases are acceptable.

(3) Specification 6.6.A.4 was added to the reporting requirements of the Administrative Controls section of the TS for both Dresden 2 and Dresden 3. This specification requires that the COLR be submitted, upon issuance, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector. The report provides the values of cycle-specific parameter limits that are applicable for the current fuel cycle. Further, these specifications require that the values of these limits be established using NRC approved methodologies and be consistent with all applicable limits of the safety analysis. The approved methodologies are the following for both Dresden 2 and Dresden 3:

(a) XN-NF-512(P)(A), "XN-3 Critical Power Correlation."

(b) XN-NF-524(P)(A), "Exxon Nuclear Critical Power Methodology for Boiling Water Reactors."

(c) XN-NF-79-71(P)(A), "Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors."

(d) XN-NF-80-19(P)(A), "Exxon Nuclear Methodology for Boiling Water Reactors."

(e) XN-NF-85-67(P)(A), "Generic Mechanical Design for Exxon Nuclear Jet Pump Boiling Water Reactors Reload Fuel."

(f) XN-NF-81-22(P)(A), "Generic Statistical Uncertainty Analysis Methodology."

Finally, the specification requires that all changes in cycle-specific parameter limits be documented in the COLR before each reload cycle or remaining part of a reload cycle and submitted upon issuance to NRC, prior to operation with the new parameter limits.

On the basis of the review of the above items, the NRC staff concludes that the licensee provided an acceptable response to those items as addressed in the NRC guidance in Generic Letter 88-16 on modifying cycle-specific parameter limits in TS. Because plant operation continues to be limited in accordance with the values of cycle-specific parameter limits that are established using NRC approved methodologies, the NRC staff concludes that these changes are administrative in nature and there is no impact on plant safety as a consequence. Accordingly, the staff finds that the proposed changes are acceptable.

As part of the implementation of Generic Letter 88-16, the staff has also reviewed a sample COLR that was provided by the licensee. On the basis of this review, the staff concludes that the format and content of the sample COLR are acceptable.

In addition to the changes identified above to implement the COLR in accordance with the guidance of Generic Letter 88-16, the licensee made a number of admininstrative changes to the Technical Specifications and Bases sections to support these changes. Since these changes, such as eliminating duplicate COLR references and updating the Bases to support the COLR change, are administrative in nature they are acceptable.

We have reviewed the request by the Commonwealth Edison Company to modify the Technical Specifications of the Dresden 2 and Dresden 3 plants that would remove the specific values of some cycle-dependent parameters from the specifications and place the values in a Core Operating Limits Report that would be referenced by the affected specifications. Based on this review, we conclude that these Technical Specification modifications are acceptable. We, also conclude that the other changes to the Technical Specifications, which support this COLR change, are administrative and are acceptable.

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