ENCLOSURE C

PROPOSED LICENSE AND TECHNICAL SPECIFICATION CHANGES

1. DRESDEN UNIT 2 (DPR-19)

2. DRESDEN UNIT 3 (DPR-25)

DRESDEN STATION UNIT 2

TECHNICAL SPECIFICATION REVISIONS

DPR-19

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Am. 34 1/30/78

2\. E. Pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of Dresden Nuclear Power Station, Units Nos. 1, 2 and 3.

3\. This license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations; 10 CFR Part 20, Section 30.34 of 10 CFR Part 30, Section 40.41 of 10 CFR Part 40, Sections 50-54 and 50-59 of 10 CFR Part 50, and Section 70.32 of 10 CFR Part 70; and is subject to the additional conditions specified below:

A. Maximum Power Level

Commonwealth Edison is authorized to operate the facility at steady state power levels not in excess of 2527 megawatts (thermal).

Am. 107 8/10/89

To be determined

B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment 107 are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

C. Reports

Commonwealth Edison shall make certain reports in accordance with the requirements of the Technical Specifications.

D. Records

Commonwealth Edison shall keep facility operating records in accordance with the requirements of the Technical Specifications.

E. For the purpose of repairing a crack in the recirculation bypass line in the "A" loop, the licensee may perform the repair program as described in a report entitled "Commonwealth Edison Company Dresden Station 2A Recirculations Pump 4" Equalizing Line Repair Program" transmitted by letter dated September 23, 1974.

List of Figures

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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. (Deleted)~~

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

Insert A

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 ~~XN-3~~ correlation. ~~(Reference XN-NF-512).~~

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

Insert A

B. <u>Core Operating Limits Report (COLR)</u> - 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.

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 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 ~~for Exxon Fuel~~ (FDLRX) - The fuel design limiting ratio ~~for Exxon fuel,~~ 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. ~~See reference XN-NF-524.~~

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 ~~XN-3~~ ANF NRC-approved critical power correlation. Refer to ~~XN-NF-524~~ Specification 6.6.A.4 for the methodology used in determining the MCPR Fuel Cladding Integrity Safety Limit.

The ~~XN-3~~ 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 ~~XN-3~~ ANF NRC-approved correlation to overpredict the number of rods in boiling transition. These conservatisms and the inherent accuracy of the ~~XN-3~~ 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.)

The ANF NRC-approved methodology listed in Specification 6.6.A.4,

ANF NRC-approved

If the reactor pressure should ever exceed the limit of applicability of the ~~XN-3~~ critical power correlation as defined in ~~XN-NF-512,~~ 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. ~~For fuel fabricated by General Electric Company, operation is further constrained to a maximum linear heat generation rate (LHGR) of 13.4 kW/ft by Specification 3.5.J. This constraint is established to provide adequate safety margin to 1% plastic strain for abnormal operational transients initiated from high power conditions. Specification 2.1.A.1 provides for equivalent safety margin for transients initiated from lower power conditions by adjusting the APRM flow-biased scram by 1/FDLRC. Specification 3.5.J establishes the maximum value of LHGR which cannot be exceeded during steady power operation for any fuel types.~~

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~~For fuel fabricated by Advanced Nuclear Fuels Corporation (ANF), ANF has performed fuel design analysis which demonstrate that fuel centerline melting point will not be reached during transient overpower condition throughout the design life of the fuel provided FDLRC is monitored. The analysis has also shown that the design criteria of 1.0% uniform cladding strain will not be exceeded during both steady state and transient operation throughout the fuel design life provided FDLRC is monitored.~~

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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

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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.