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

May 23, 1986

Docket No. 50-397

Mr. G. C. Sorensen, Manager
Regulatory Programs
Washington Public Power Supply System
P.O. Box 968
3000 George Washington Way
Richland, Washington 99352

Dear Mr. Sorensen:

Subject: Issuance of Amendment No. 28 to Facility Operating License NPF-21 - WPPSS Nuclear Project No. 2

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 28 to Facility Operating License NPF-21 to the Washington Public Power Supply System for WPPSS Nuclear Project No. 2, located in Benton County near Richland, Washington. This amendment is in response to your letters dated February 26, April 7, 24, 25, 20, and May 22, 1986.

This amendment revises the WNP-2 Technical Specifications to support the operation of WNP-2 at full rated power during the next fuel cycle, Cycle 2. This reload amendment changes the Technical Specifications in the following areas: (1) establishes operating limits for all fuel types for the upcoming Cycle 2 operation; (2) reflects the replacement of 128 initial core fuel assemblies with Exxon Nuclear Company (ENC) fuel assemblies. Also included are related modifications of the Bases section of the Technical Specification to account for the use of Exxon fuel assemblies.

A copy of the related safety evaluation supporting Amendment No. 28 to Facility Operating License No. NPF-21 is enclosed.

Sincerely,

Elinor G. Adensam

Elinor G. Adensam, Director
BWR Project Directorate No. 3
Division of BWR Licensing

Enclosures:
1. Amendment No. 28 to Facility Operating License No. NPF-21
2. Safety Evaluation

cc w/enclosures:
See next page

Mr. G. C. Sorensen, Manager
Washington Public Power Supply System

WPPSS Nuclear Project No. 2
(WNP-2)

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

WASHINGTON PUBLIC POWER SUPPLY SYSTEM

DOCKET NO. 50-397

WPPSS NUCLEAR PROJECT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 28
License No. NPF-21

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

   A. The application for amendment filed by the Washington Public Power Supply System (the Supply System, also the licensee), dated February 26, 1986, as supplemented on April 7, 24, 25, 30 and May 22, 1986, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's 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 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 set forth in 10 CFR Chapter I;

   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 enclosure to this license amendment and paragraph 2.C.(2) of the Facility Operating License No. NPF-21 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 28, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Elinor G. Adensam

Elinor G. Adensam, Director
BWR Project Directorate No. 3
Division of BWR Licensing

Enclosure:
Changes to the Technical
Specifications

Date of Issuance: May 23, 1986

ENCLOSURE TO LICENSE AMENDMENT NO. 28

FACILITY OPERATING LICENSE NO. NPF-21

DOCKET NO. 50-397

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain a vertical line indicating the area of change.

REMOVE	INSERT
i	i
ii	ii
iii	iii
xx	xx
	xx(a)
xxi	xxi
xxiv	xxiv
1-1	1-1
1-2	1-2
1-3	1-3
1-4	1-4
1-5	1-5
1-6	1-6
1-7	1-7
1-8	1-8
B 2-1	B 2-1
B 2-2	B 2-2
B 2-3	B 2-3
B 2-4	
3/4 1-2	3/4 1-2
3/4 1-8	3/4 1-8
3/4 2-1	3/4 2-1
3/4 2-2	3/4 2-2
3/4 2-3	3/4 2-3
3/4 2-4	3/4 2-4
	3/4 2-4A
	3/4 2-4B
	3/4 2-4C
3/4 2-6	3/4 2-6
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	3/4 2-9
	3/4 2-10
3/4 3-41	3/4 3-41
B 3/4 1-1	B 3/4 1-1
B 3/4 1-2	B 3/4 1-2
B 3/4 1-3	B 3/4 1-3
B 3/4 1-4	B 3/4 1-4
B 3/4 2-1	B 2/4 2-1
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1.0 DEFINITIONS

The following terms are defined so that uniform interpretation of these specifications may be achieved. The defined terms appear in capitalized type and shall be applicable throughout these Technical Specifications.

ACTION

1.1 ACTION shall be that part of a Specification which prescribes remedial measures required under designated conditions.

AVERAGE BUNDLE EXPOSURE

1.2 The AVERAGE BUNDLE EXPOSURE is equal to the sum of the axially averaged exposure of all the fuel rods in the specified bundle divided by the number of fuel rods in the bundle.

AVERAGE PLANAR EXPOSURE

1.3 The AVERAGE PLANAR EXPOSURE shall be applicable to a specific planar height and is equal to the sum of the exposure of all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle.

AVERAGE PLANAR LINEAR HEAT GENERATION RATE

1.4 The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) shall be applicable to a specific planar height and is equal to the sum of the LINEAR HEAT GENERATION RATES for all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle.

CHANNEL CALIBRATION

1.5 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel including the sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK

1.6 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

CHANNEL FUNCTIONAL TEST

1.7 A CHANNEL FUNCTIONAL TEST shall be:

a. Analog channels - the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY including alarm and/or trip functions and channel failure trips.

b. Bistable channels - the injection of a simulated signal into the sensor to verify OPERABILITY including alarm and/or trip functions.

The CHANNEL FUNCTIONAL TEST may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is tested.

CORE ALTERATION

1.8 CORE ALTERATION shall be the addition, removal, relocation or movement of fuel, sources, incore instruments or reactivity controls within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of the movement of a component to a safe conservative position.

CRITICAL POWER RATIO

1.9 The CRITICAL POWER RATIO (CPR) shall be that power in the assembly which is calculated by application of the XN-3 correlation to cause some point in the assembly to experience boiling transition divided by the actual assembly operating power.

DOSE EQUIVALENT I-131

1.10 DOSE EQUIVALENT I-131 shall be that concentration of I-131, microcuries per 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."

$\bar{E}$-AVERAGE DISINTEGRATION ENERGY

1.11 $\bar{E}$ shall be the average, weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling, of the sum of the average beta and gamma energies per disintegration, in MeV, for isotopes, with half-lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME

1.12 The EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ECCS actuation setpoint at the channel sensor until the ECCS equipment is capable of performing its safety function, i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc. Times shall include diesel generator starting and sequence loading delays where applicable. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME

1.13 The END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME shall be that time interval to energization of the recirculation pump circuit breaker trip coil from when the monitored parameter exceeds its trip setpoint at the channel sensor of the associated:

a. Turbine throttle valves channel sensor contact opening, and

b. Turbine governor valves initiation of valve fast closure.

The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

FRACTION OF LIMITING POWER DENSITY

1.14 The FRACTION OF LIMITING POWER DENSITY (FLPD) shall be the LHGR existing at a given location divided by the specified LHGR limit for that bundle type.

FRACTION OF RATED THERMAL POWER

1.15 The FRACTION OF RATED THERMAL POWER (FRTP) shall be the measured THERMAL POWER divided by the RATED THERMAL POWER.

FREQUENCY NOTATION

1.16 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.1.

GASEOUS RADWASTE TREATMENT SYSTEM

1.17 A GASEOUS RADWASTE TREATMENT SYSTEM shall be any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

IDENTIFIED LEAKAGE

1.18 IDENTIFIED LEAKAGE shall be:

a. Leakage into collection systems, such as pump seal or valve packing leaks, that is captured and conducted to a sump or collecting tank, or

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of the leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE.

ISOLATION SYSTEM RESPONSE TIME

1.19 The ISOLATION SYSTEM RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its isolation actuation setpoint at the channel sensor until the isolation valves travel to their required positions. Times shall include diesel generator starting and sequence loading delays where applicable. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

LIMITING CONTROL ROD PATTERN

1.20 A LIMITING CONTROL ROD PATTERN shall be a pattern which results in the core being on a thermal hydraulic limit, i.e., operating on a limiting value for APLHGR, LHGR, or MCPR.

LINEAR HEAT GENERATION RATE

1.21 LINEAR HEAT GENERATION RATE (LHGR) shall be the heat generation per unit length of fuel rod. It is the integral of the heat flux over the heat transfer area associated with the unit length.

LOGIC SYSTEM FUNCTIONAL TEST

1.22 A LOGIC SYSTEM FUNCTIONAL TEST shall be a test of all logic components, i.e., all relays and contacts, all trip units, solid state logic elements, etc, of a logic circuit, from sensor through and including the actuated device, to verify OPERABILITY. The LOGIC SYSTEM FUNCTIONAL TEST may be performed by any series of sequential, overlapping or total system steps such that the entire logic system is tested.

MAXIMUM FRACTION OF LIMITING POWER DENSITY

1.23 The MAXIMUM FRACTION OF LIMITING POWER DENSITY (MFLPD) shall be highest value of the FLPD which exists in the core.

MAXIMUM TOTAL PEAKING FACTOR

1.24 The MAXIMUM TOTAL PEAKING FACTOR (MTPF) shall be the largest TPF which exists in the core for a given class of fuel for a given operating condition.

MEMBER(S) OF THE PUBLIC

1.25 MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the utility, its contractors or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational or other purposes not associated with the plant.

MINIMUM CRITICAL POWER RATIO

1.26 The MINIMUM CRITICAL POWER RATIO (MCPR) shall be the smallest CPR which exists in the core.

OFFSITE DOSE CALCULATION MANUAL

1.27 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the current methodology and parameters used in the calculation of offsite doses due to radioactive gaseous and liquid effluents in the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints and in the conduct of the environmental radiological monitoring program.

OPERABLE - OPERABILITY

1.28 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s) and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

OPERATIONAL CONDITION - CONDITION

1.29 An OPERATIONAL CONDITION, i.e., CONDITION, shall be any one inclusive combination of mode switch position and average reactor coolant temperature as specified in Table 1.2.

PHYSICS TESTS

1.30 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation as (1) described in Chapter 14 of the FSAR, (2) authorized under the provisions of 10 CFR 50.59, or (3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

1.31 PRESSURE BOUNDARY LEAKAGE shall be leakage through a non-isolable fault in a reactor coolant system component body, pipe wall, or vessel wall.

PRIMARY CONTAINMENT INTEGRITY

1.32 PRIMARY CONTAINMENT INTEGRITY shall exist when:

a. All primary containment penetrations required to be closed during accident conditions are either:

   1. Capable of being closed by an OPERABLE primary containment automatic isolation system, or

   2. Closed by at least one manual valve, blind flange, or deactivated automatic valve secured in its closed position, except as provided in Table 3.6.3-1 of Specification 3.6.3.

b. All primary containment equipment hatches are closed and sealed.

c. Each primary containment air lock is in compliance with the requirements of Specification 3.6.1.3.

d. The primary containment leakage rates are within the limits of Specification 3.6.1.2.

e. The suppression chamber is in compliance with the requirements of Specification 3.6.2.1.

f. The sealing mechanism associated with each primary containment penetration; e.g., welds, bellows, or O-rings, is OPERABLE.

PROCESS CONTROL PROGRAM

1.33 The PROCESS CONTROL PROGRAM (PCP) shall contain the sampling, analysis, and formulation determination by which SOLIDIFICATION of radioactive wastes from liquid systems is assured.

PURGE - PURGING

1.34 PURGE or PURGING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

RATED THERMAL POWER

1.35 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3323 MWt.

REACTOR PROTECTION SYSTEM RESPONSE TIME

1.36 REACTOR PROTECTION SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until deenergization of the scram pilot valve solenoids. The response time may be measured by any series of sequential, overlapping, or total steps such that the entire response time is measured.

REPORTABLE EVENT

1.37 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

ROD DENSITY

1.38 ROD DENSITY shall be the number of control rod notches inserted as a fraction of the total number of control rod notches. All rods fully inserted is equivalent to 100% ROD DENSITY.

SECONDARY CONTAINMENT INTEGRITY

1.39 SECONDARY CONTAINMENT INTEGRITY shall exist when:

a. All secondary containment penetrations required to be closed during accident conditions are either:

   1. Capable of being closed by an OPERABLE secondary containment automatic isolation system, or

   2. Closed by at least one manual valve, blind flange, or deactivated automatic valve secured in its closed position.

b. All secondary containment hatches and blowout panels are closed and sealed.

c. The standby gas treatment system is in compliance with the requirements of Specification 3.6.5.3.

SECONDARY CONTAINMENT INTEGRITY (Continued)

d. At least one door in each access to the secondary containment is closed.

e. The sealing mechanism associated with each secondary containment penetration, e.g., welds, bellows, or O-rings, is OPERABLE.

f. The pressure within the secondary containment is less than or equal to the value required by Specification 4.6.5.1.a.

SHUTDOWN MARGIN

1.40 SHUTDOWN MARGIN shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming all control rods are fully inserted except for the single control rod of highest reactivity worth which is assumed to be fully withdrawn and the reactor is in the shutdown condition; cold, i.e., 68°F; and xenon free.

SITE BOUNDARY

1.41 The SITE BOUNDARY shall be that line beyond which the land is not owned, leased, or otherwise controlled by the licensee.

SOLIDIFICATION

1.42 SOLIDIFICATION shall be the conversion of radioactive wastes from liquid systems to a homogeneous (uniformly distributed), monolithic, immobilized solid with definite volume and shape, bounded by a stable surface of distinct outline on all sides (free-standing).

SOURCE CHECK

1.43 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a radioactive source.

STAGGERED TEST BASIS

1.44 A STAGGERED TEST BASIS shall consist of:

a. A test schedule for n systems, subsystems, trains, or other designated components obtained by dividing the specified test interval into n equal subintervals.

b. The testing of one system, subsystem, train, or other designated component at the beginning of each subinterval.

THERMAL POWER

1.45 THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

TOTAL PEAKING FACTOR

1.46 The TOTAL PEAKING FACTOR (TPF) shall be the ratio of local LHGR for any specific location on a fuel rod divided by the core average LHGR associated with the fuel bundles of the same type operating at the core average bundle power.

TURBINE BYPASS SYSTEM RESPONSE TIME

1.47 The TURBINE BYPASS SYSTEM RESPONSE TIME shall be that time interval from when the turbine bypass control unit generates a turbine bypass valve flow signal until the turbine bypass valves travel to their required positions. The response time may be measured by any series of sequential, overlapping, or total steps such that the entire response time is measured.

UNIDENTIFIED LEAKAGE

1.48 UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE.

UNRESTRICTED AREA

1.49 An UNRESTRICTED AREA shall be any area at or beyond the SITE BOUNDARY access to which is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials, or any area within the site boundary used for residential quarters or for industrial, commercial, institutional, and/or recreational purposes.

VENTILATION EXHAUST TREATMENT SYSTEM

1.50 A VENTILATION EXHAUST TREATMENT SYSTEM shall be any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal adsorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment (such a system is not considered to have any effect on noble gas effluents). Engineered Safety Features (ESF) atmospheric cleanup systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.

VENTING

1.51 VENTING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration, or other operating condition, in such a manner that replacement air or gas is not provided or required during VENTING. Vent, used in system names, does not imply a VENTING process.

BASES

INTRODUCTION

The fuel cladding, reactor pressure vessel and primary system piping are the principal barriers to the release of radioactive materials to the environs. Safety Limits are established to protect the integrity of these barriers during normal plant operations and anticipated transients. The fuel cladding integrity Safety Limit is set such that no fuel damage is calculated to occur if the limit is not violated. Because fuel damage is not directly observable, a step-back approach is used to establish a Safety Limit such that the MCPR is not less than 1.06 for two recirculation loop operation and 1.07 for single recirculation loop operation for both GE and ENC fuel. MCPR greater than 1.06 for two recirculation loop operation and 1.07 for single recirculation loop operation represents a conservative margin relative to the conditions required to maintain fuel cladding integrity. The fuel cladding is one of the physical barriers which separate the radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions and the Limiting Safety System Settings. While fission product migration from cladding perforation is just as measurable as that from use related cracking, the thermally caused cladding perforations signal a threshold beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding integrity Safety Limit is defined with a 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 percent of the fuel rods in the core do not experience transition boiling (Reference XN-NF-524 (A), Rev. 1).

2.1 SAFETY LIMITS

2.1.1. THERMAL POWER, Low Pressure or Low Flow

For certain conditions of pressure and flow, the XN-3 correlation is not valid for all critical power calculations. The XN-3 correlation is not valid for bundle mass velocities less than $.25 \times 10^6$ lbs/hr-ft^2 or pressures less than 585 psig. Therefore, the fuel cladding integrity Safety Limit is established by other means. This is done by establishing a limiting condition on core THERMAL POWER with the following basis. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be greater than 4.5 psi. Analyses show that with a bundle flow of 28×10^3 lbs/h (approximately a mass velocity of $.25 \times 10^6$ lbs/hr-ft^2), bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.5 psi driving head will be greater than 28×10^3 lbs/h. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power

THERMAL POWER, Low Pressure or Low Flow (Continued)

at this flow is approximately 3.35 MWt. With the design peaking factors, this corresponds to a THERMAL POWER of more than 50% of RATED THERMAL POWER. Thus, a THERMAL POWER limit of 25% of RATED THERMAL POWER for reactor pressure below 585 psig is conservative.

2.1.2 THERMAL POWER, High Pressure and High Flow

The fuel cladding integrity Safety Limit is set such that no fuel damage is calculated to occur if the limit is not violated. Since the parameters which result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions resulting in a departure from nucleate boiling have been used to mark the beginning of the region where fuel damage could occur. Although it is recognized that a departure from nucleate boiling would not necessarily result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. However, the uncertainties in monitoring the core operating state and in the procedures used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore, the fuel cladding integrity Safety Limit is defined as the CPR in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition considering the power distribution within the core and all uncertainties.

The Safety Limit MCPR is determined using the Exxon Nuclear Critical Power Methodology for boiling water reactors[a] which is a statistical model that combines all of the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the Exxon nuclear critical heat flux-enthalpy XN-3 correlation. The XN-3 correlation is valid over the range of conditions used in the tests of the data used to develop the correlation.

The required input to the statistical model are the uncertainties listed in Bases Table B2.1.2-1 and the nominal values of the core parameters listed in Bases Table B2.1.2-2.

The bases for the uncertainties in the core parameters are given in XN-NF-524(A), Rev. 1[a] and the basis for the uncertainty in the XN-3 correlation is given in XN-FN-512(A), Rev. 1[b]. The power distribution is based on a typical 764 assembly core in which the rod pattern was arbitrarily chosen to produce a skewed power distribution having the greatest number of assemblies at the highest power levels. The worst distribution during any fuel cycle would not be as severe as the distribution used in the analysis.

a. Exxon Nuclear Critical Power Methodology for Boiling Water Reactors, XN-NF-524(A), Rev. 1.

b. Exxon Nuclear Company XN-3 Critical Power Correlation, XN-NF-512(a), Rev. 1.

BASES TABLE B2.1.2-1

UNCERTAINTIES CONSIDERED IN THE MCPR SAFETY LIMIT

Parameter	STANDARD DEVIATION*
Feedwater Flow Rate	.0176
Feedwater Temperature	.0076
Core Pressure	.0050
Total Core Flow Rate	.0250
Core Inlet Enthalpy	.0024
XN-3 Critical Power Correlation	.0411
Assembly Flow Rate	.0280
Power Distribution:	
Radial Peaking Factor	.0528
Local Peaking Factor	.0246

* Fraction of Nominal Value.

3/4.1.2 REACTIVITY ANOMALIES

LIMITING CONDITION FOR OPERATION

3.1.2 The reactivity difference between the monitored core k_{eff} and the predicted core k_{eff} shall not exceed 1% delta k/k.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With the reactivity equivalence difference exceeding 1% delta k/k:

a. Within 12 hours perform an analysis to determine and explain the cause of the reactivity difference; operation may continue if the difference is explained and corrected.

b. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.1.2 The reactivity difference between the monitored core k_{eff} and the predicted core k_{eff} shall be verified to be less than or equal to 1% delta k/k:

a. During the first startup following CORE ALTERATIONS, and

b. At least once per 31 effective full power days.

FOUR CONTROL ROD GROUP SCRAM INSERTION TIMES

LIMITING CONDITION FOR OPERATION

3.1.3.4 The average scram insertion time, from the fully withdrawn position, for the four control rods arranged in a two-by-two array, based on deenergization of the scram pilot valve solenoids as time zero, shall not exceed any of the following:

Position Inserted From Fully Withdrawn	Average Scram Insertion Time (Seconds)
45	0.455
39	0.920
25	2.052
5	3.706

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

a. With the average scram insertion times of control rods exceeding the above limits:

   1. Declare the control rods with the slower than average scram insertion times inoperable until an analysis is performed to determine that required scram reactivity remains for the slow four control rod group, and

   2. Perform the Surveillance Requirements of Specification 4.1.3.2.c at least once per 60 days when operation is continued with an average scram insertion time(s) in excess of the average scram insertion time limit.

   Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.1.3.4 All control rods shall be demonstrated OPERABLE by scram time testing from the fully withdrawn position as required by Surveillance Requirement 4.1.3.2.

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.1 All AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGRs) for each type of fuel as a function of AVERAGE PLANAR EXPOSURE for GE fuel and average bundle exposure for ENC fuel shall not exceed the limits shown in Figures 3.2.1-1, 3.2.1-2, and 3.2.1-3. The limits for single loop operation are shown in Figures 3.2.1-4, 3.2.1-5, and 3.2.1-6.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

ACTION:

With an APLHGR exceeding the limits of Figure 3.2.1-1, 3.2.1-2, or 3.2.1-3, initiate corrective action within 15 minutes and restore APLHGR to within the required limits within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

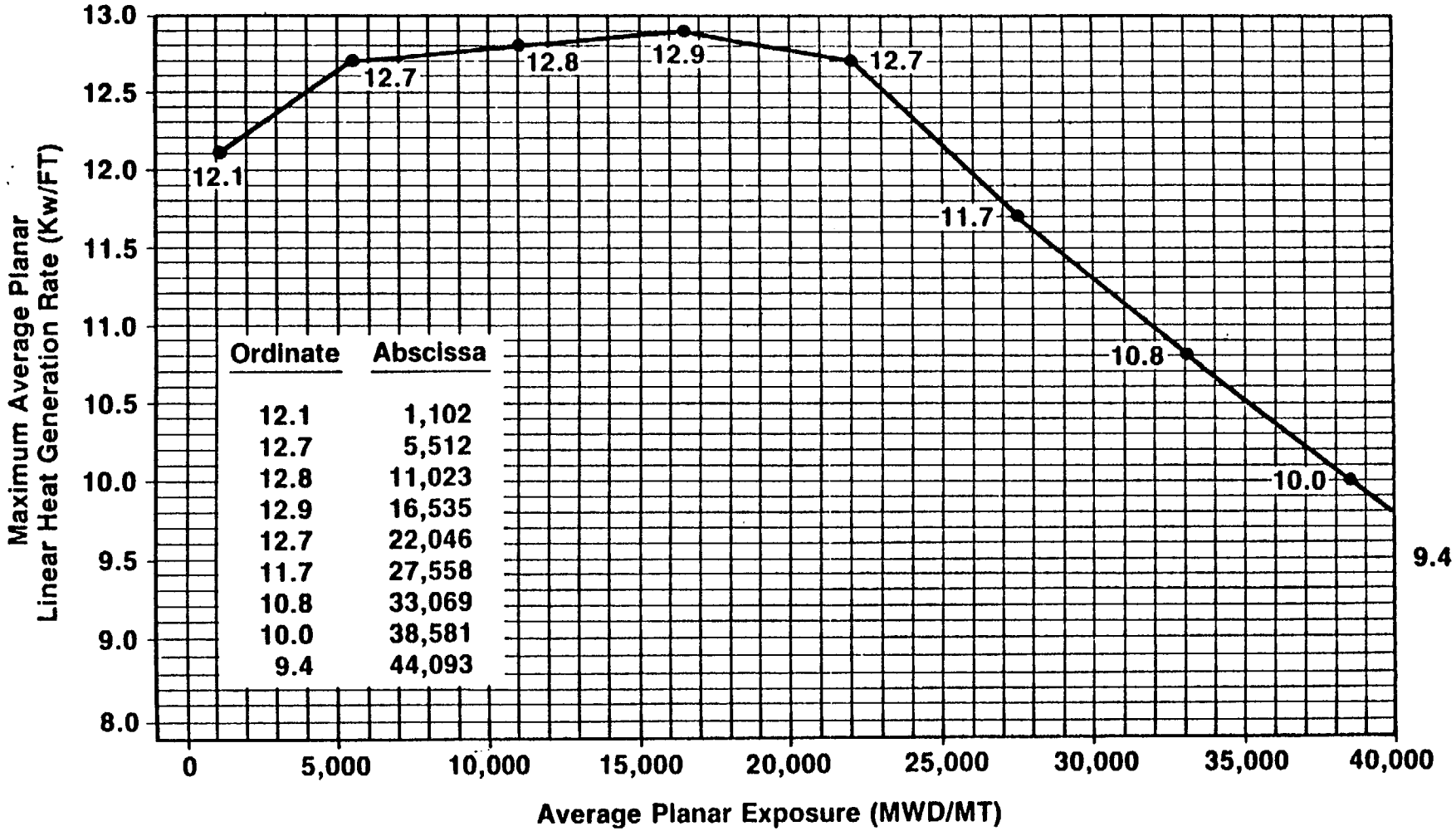
SURVEILLANCE REQUIREMENTS

4.2.1 All APLHGRs shall be verified to be equal to or less than the limits determined from Figures 3.2.1-1, 3.2.1-2, and 3.2.1-3:

a. At least once per 24 hours,

b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

c. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for APLHGR.

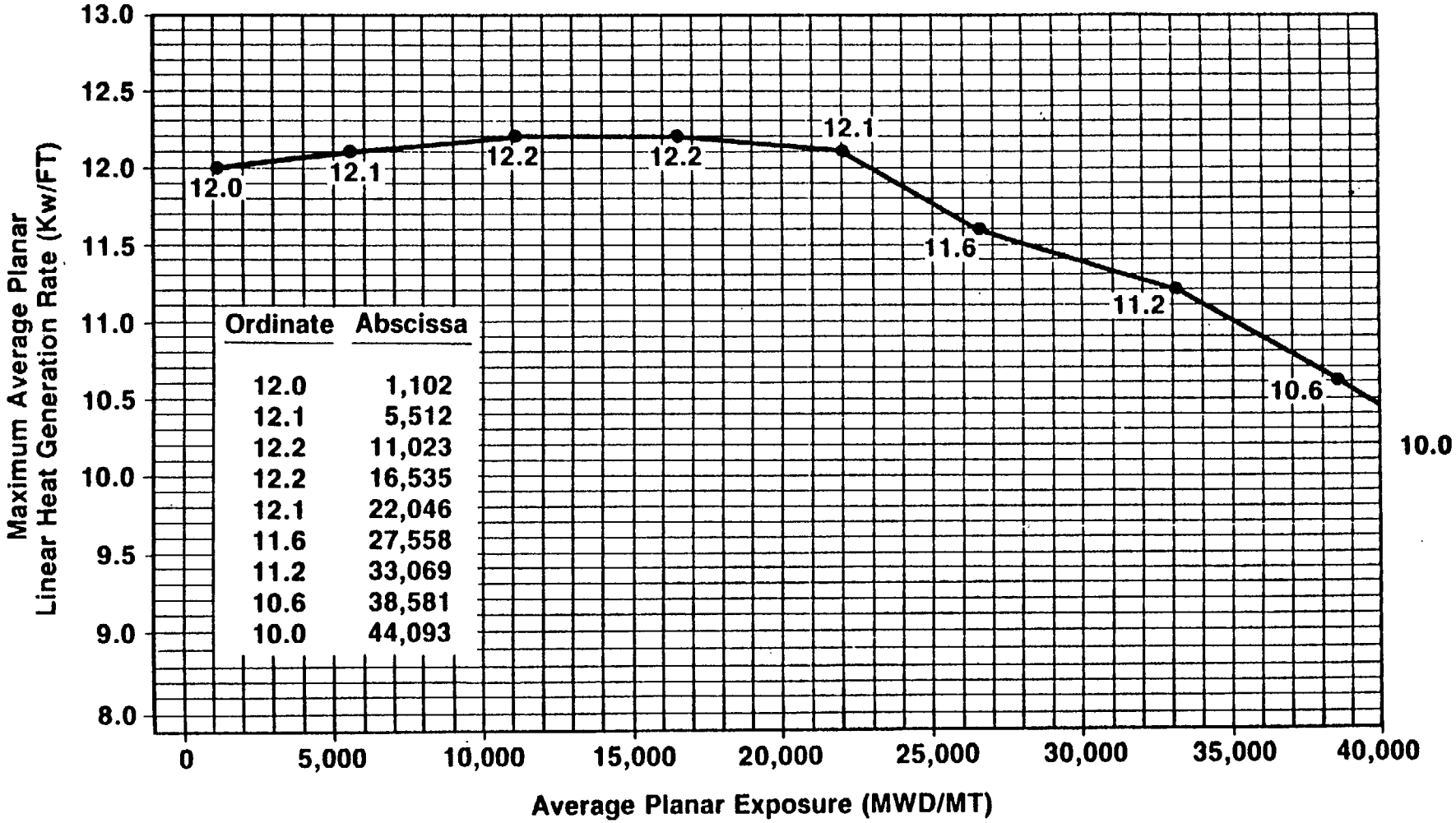


Ordinate	Abscissa
12.1	1,102
12.7	5,512
12.8	11,023
12.9	16,535
12.7	22,046
11.7	27,558
10.8	33,069
10.0	38,581
9.4	44,093

Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) Versus Average Planar Exposure Initial Core Fuel Type 8CR183

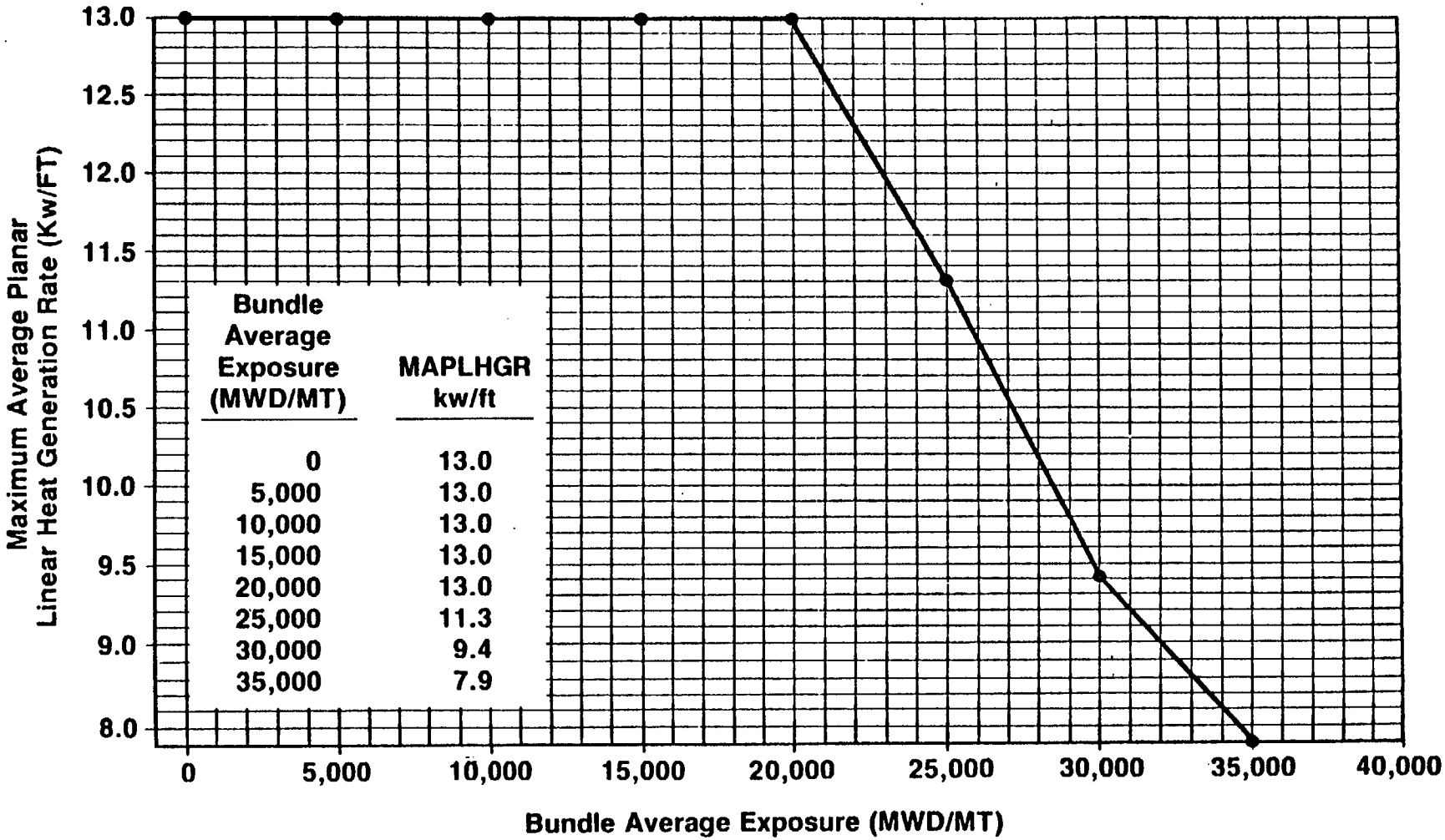
Figure 3.2.1-1

860599.4A



Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) Versus Average Planar Exposure Initial Core Fuel Type 8CR233
Figure 3.2.1-2

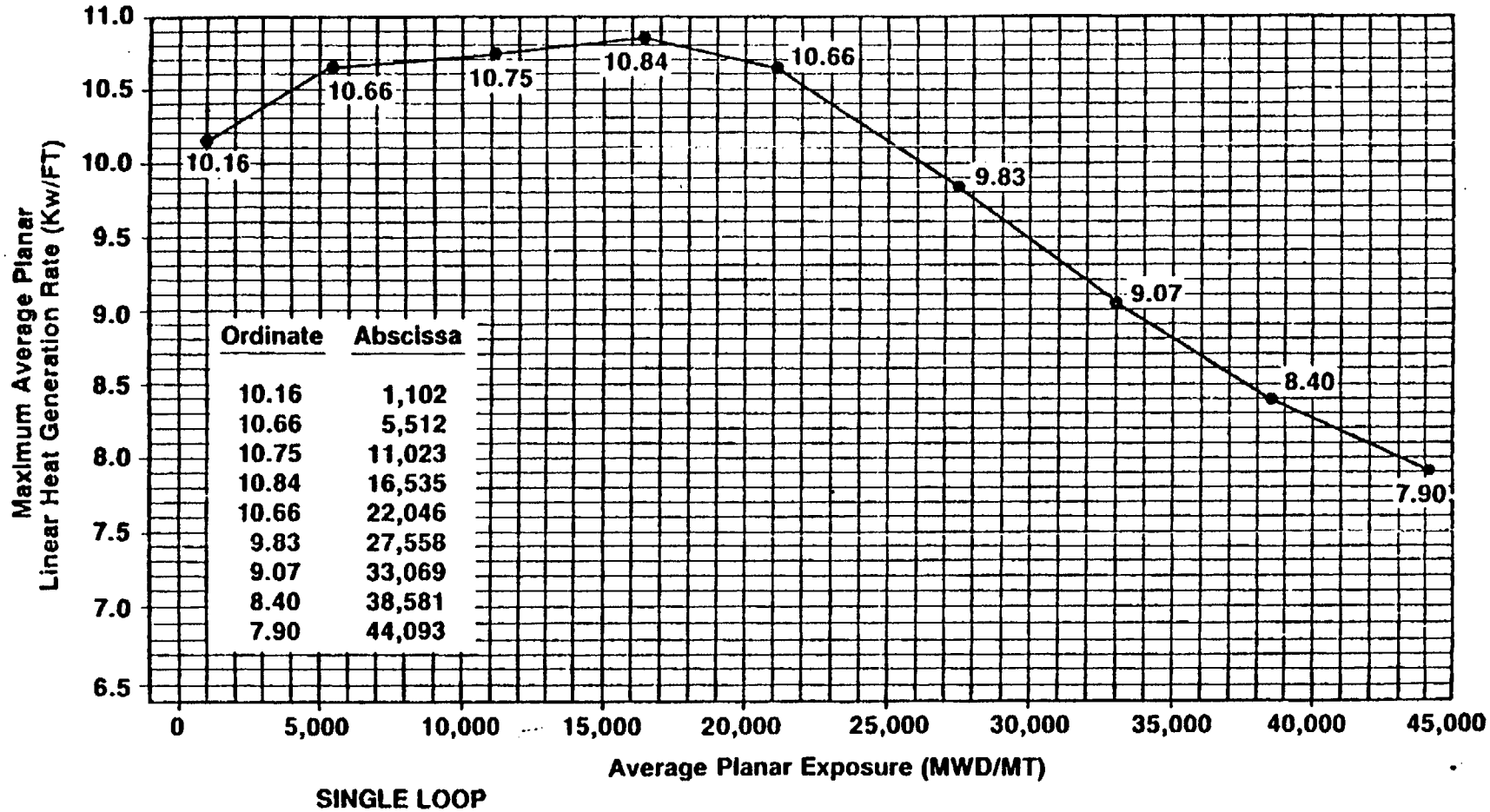
860599.5A



Bundle Average Exposure (MWD/MT)	MAPLHGR kw/ft
0	13.0
5,000	13.0
10,000	13.0
15,000	13.0
20,000	13.0
25,000	11.3
30,000	9.4
35,000	7.9

Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) Versus Bundle Average Exposure
ENC XN-1 Fuel
Figure 3.2.1-3

860599.2A

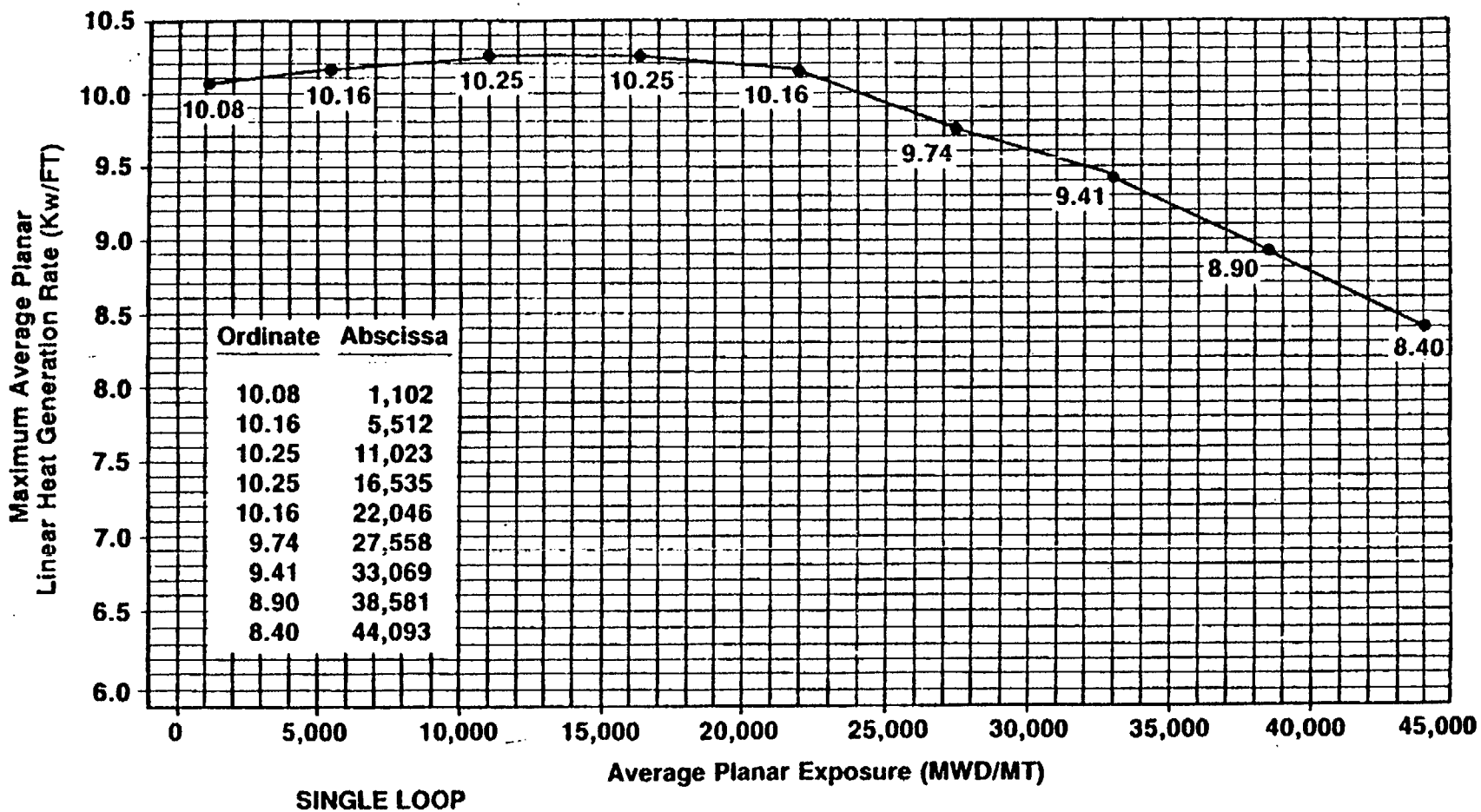


SINGLE LOOP

Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) Versus Average Planar Exposure Initial Core Fuel Type 8CR183

Figure 3.2.1-4

860599.5A

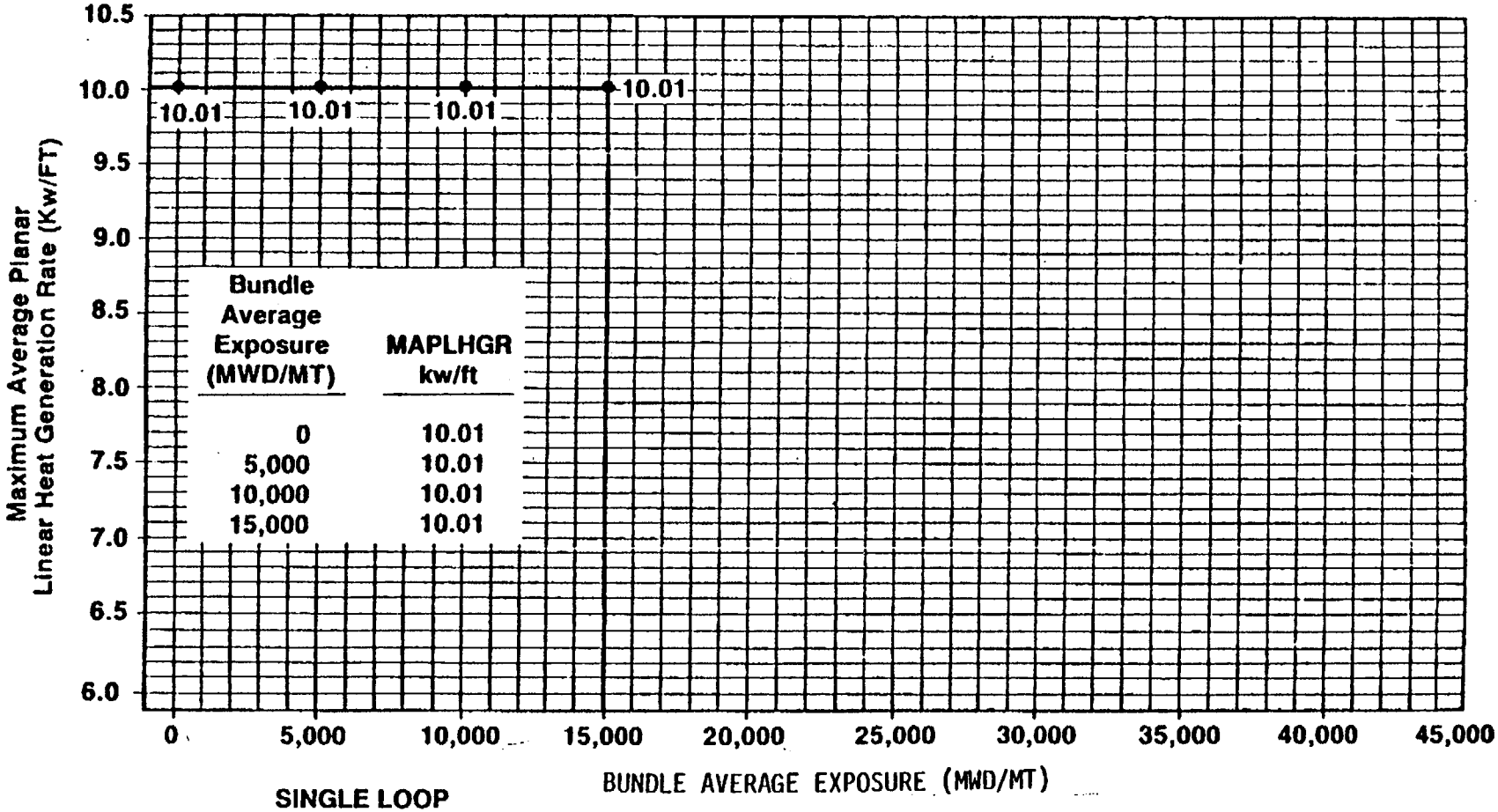


SINGLE LOOP

Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) Versus Average Planar Exposure Initial Core Fuel Type 8CR233

Figure 3.2.1-5

660599.5A



Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) Versus
BUNDLE AVERAGE EXPOSURE
ENC XN-1 FUEL
Figure 3.2.1-6

660599.5A

3/4.2.3 MINIMUM CRITICAL POWER RATIO

LIMITING CONDITION FOR OPERATION

3.2.3 The MINIMUM CRITICAL POWER RATIO (MCPR) shall be:

a. Greater than or equal to the applicable MCPR limit determined from Table 3.2.3-1 during steady state operation at rated core flow, or

b. Greater than or equal to the greater of the two values determined from Table 3.2.3-1 and Figure 3.2.3-1 during steady state operation at other than rated core flow.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25 percent of RATED THERMAL POWER.

ACTION: With MCPR less than the applicable MCPR limit determined from Table 3.2.3-1 and Figure 3.2.3-1, initiate corrective action within 15 minutes and restore MCPR to within the required limit within 2 hours or reduce THERMAL POWER to less than 25 percent of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.3.1 MCPR shall be determined to be greater than or equal to the applicable MCPR limit determined from Table 3.2.3-1 and Figure 3.2.3-1.

a. At least once per 24 hours,

b. Within 12 hours after completion of a THERMAL POWER increase of at least 15 percent of RATED THERMAL POWER, and

c. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for MCPR.

Table 3.2.3-1

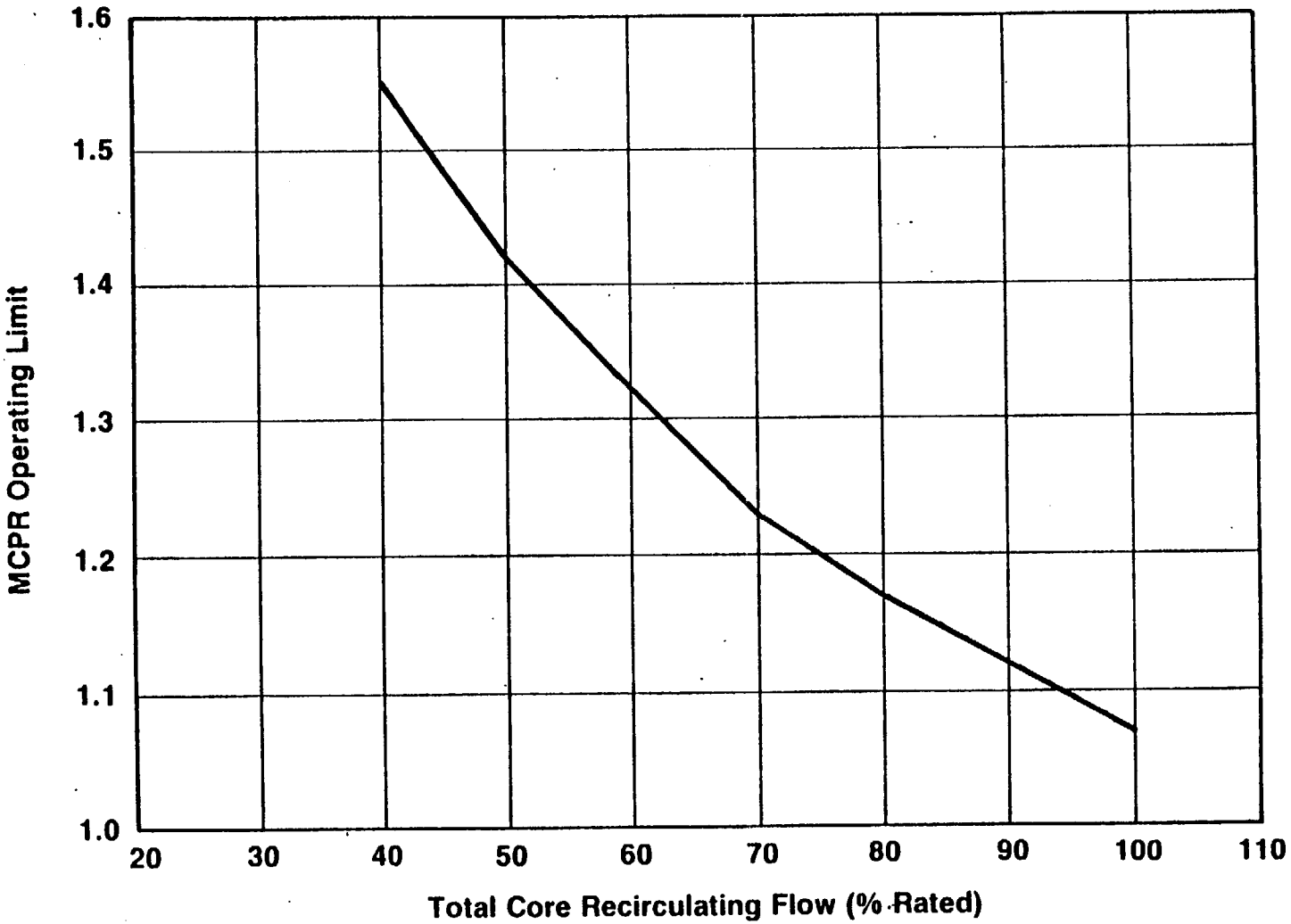
MCPR OPERATING LIMITS FOR

RATED CORE FLOW

Equipment Status	MCPR Operating Limit 100% Core Flow	MCPR Operating Limit 106% Core Flow
1. Normal*	1.27 ENC Fuel 1.28 GE Fuel	1.27 ENC Fuel 1.28 GE Fuel
2. Control Rod Insertion Bounded by Tech. Spec. Limits (3.1.3.4 - p3/4 1-7)	1.32 Both Fuel Types	1.32 Both Fuel Types
3. RTP Inoperable, Normal Scram	1.32 ENC Fuel 1.33 GE Fuel	1.33 ENC Fuel 1.34 GE Fuel

* This MCPR is based on the ENC reload safety analyses performed using the control rod insertion times shown below (defined as normal scram). In the event that surveillance 4.1.3.2 shows these scram insertion times may be exceeded, the plant thermal limits of Step 1. above are to default to the values in Step 2. above and the scram insertion times must meet the requirements of Tech. Spec. 3.1.3.4.

Position Inserted From Fully Withdrawn	Slowest Measured Average Control Rod Insertion Time to Specified Notches for Each Group of 4 Control Rods Arranged in a Two-by-Two Array (Seconds)
Notch 45	.404
Notch 39	.660
Notch 25	1.504
Notch 5	2.624



Reduced Flow MCPR Operating Limit

Figure 3.2.3-1

860599.1A

3/4.2.4 LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.4 The LINEAR HEAT GENERATION RATE (LHGR) for GE fuel shall not exceed 13.4 kW/ft. The LHGR for ENC fuel shall not exceed the values shown in Figure 3.2.4-1.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

ACTION:

With the LHGR of any fuel rod exceeding the limit, initiate corrective action within 15 minutes and restore the LHGR to within the limit within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

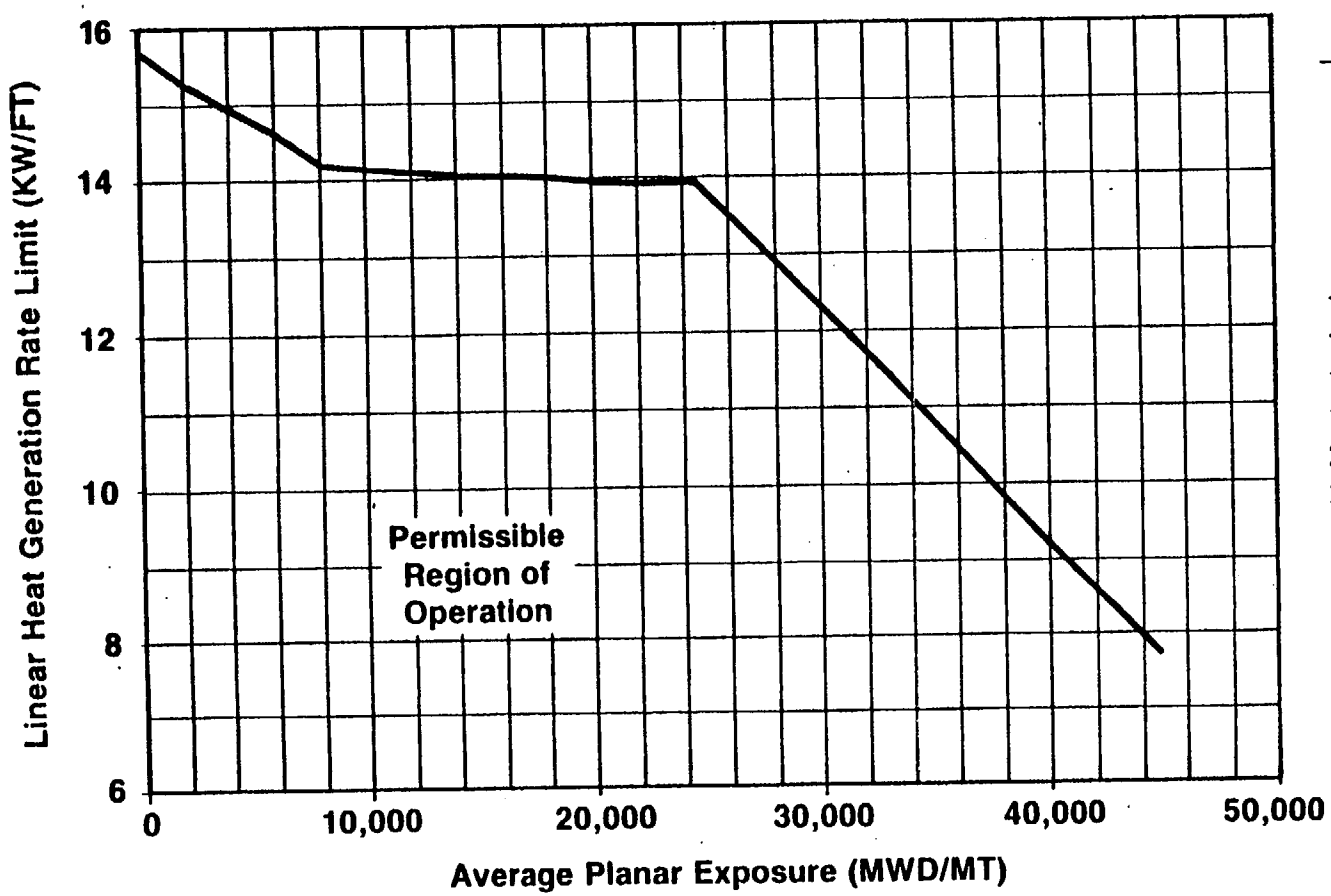
SURVEILLANCE REQUIREMENTS

4.2.4 LHGRs shall be determined to be equal to or less than the limit:

a. At least once per 24 hours,

b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

c. Initially and at least once per 12 hours when the reactor is operating on a LIMITING CONTROL ROD PATTERN for LHGR.



EXP	LHGR
0	15.62
510	15.621
2,580	15.10
5,230	14.71
7,940	14.19
10,470	14.13
13,220	14.06
15,990	14.06
18,708	14.00
21,590	13.93
24,420	13.93
27,280	13.08
30,150	12.24
33,050	11.40
35,960	10.47
38,900	9.55
41,830	8.65
44,760	7.77

Linear Heat Generation Rate (LHGR) Limit
Versus Average Planar Exposure
Exxon 8 × 8 Fuel

Figure 3.2.4-1

860599.3A

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.4.2 The end-of-cycle recirculation pump trip (EOC-RPT) system instrumentation channels shown in Table 3.3.4.2-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.4.2-2 and with the END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME as shown in Table 3.3.4.2-3.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to (30)% of RATED THERMAL POWER.

ACTION:

a. With an end-of-cycle recirculation pump trip system instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.4.2-2, declare the channel inoperable until the channel is restored to OPERABLE status with the channel setpoint adjusted consistent with the Trip Setpoint value.

b. With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels per Trip System requirement for one or both trip systems, place the inoperable channel(s) in the tripped condition within one hour.

c. With the number of OPERABLE channels two or more less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system and:

   1. If the inoperable channels consist of one turbine governor valve channel and one turbine throttle valve channel, place both inoperable channels in the tripped condition within one hour.

   2. If the inoperable channels include two turbine governor valve channels or two turbine throttle valve channels, declare the trip system inoperable.

d. With one trip system inoperable, restore the inoperable trip system to OPERABLE status within 72 hours or evaluate MCPR to be equal to or greater than the applicable MCPR limit without EOC-RPT within one hour* or take the ACTION required by Specification 3.2.3.

e. With both trip systems inoperable, restore at least one trip system to OPERABLE status within one hour or evaluate MCPR to be equal to or greater than the applicable MCPR limit without EOC-RPT within one hour* or take the ACTION required by Specification 3.2.3.

---

*If MCPR is evaluated to be equal to or greater than the applicable MCPR limit without EOC-RPT within one hour, operation may continue and the provisions of Specification 3.0.4 are not applicable.

3/4.1.1 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that (1) the reactor can be made subcritical from all operating conditions, (2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and (3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

Since core reactivity values will vary through core life as a function of fuel depletion and poison burnup, the demonstration of SHUTDOWN MARGIN will be performed in the cold, xenon-free condition and shall show the core to be subcritical by at least R + 0.38% delta k/k or R + 0.28% delta k/k, as appropriate. The value of R in units of % delta k/k is the difference between the calculated value of maximum core reactivity during the operating cycle and the calculated beginning-of-life core reactivity. The value of R must be positive or zero and must be determined for each fuel loading cycle.

Two different values are supplied in the Limiting Condition for Operation to provide for the different methods of demonstration of the SHUTDOWN MARGIN. The highest worth rod may be determined analytically or by test. The SHUTDOWN MARGIN is demonstrated by an insequence control rod withdrawal at the beginning of life fuel cycle conditions, and, if necessary, at any future time in the cycle if the first demonstration indicates that the required margin could be reduced as a function of exposure. Observation of subcriticality in this condition assures subcriticality with the most reactive control rod fully withdrawn.

This reactivity characteristic has been a basic assumption in the analysis of plant performance and can be best demonstrated at the time of fuel loading, but the margin must also be determined anytime a control rod is incapable of insertion.

3/4.1.2 REACTIVITY ANOMALIES

Since the SHUTDOWN MARGIN requirement is small, a careful check on actual reactor conditions compared to the predicted conditions is necessary. Any changes in reactivity from that predicted (predicted core K_{eff}) can be determined from the core monitoring system (monitored core K_{eff}). In the absence of any deviation in plant operating conditions of reactivity anomaly, these values should be essentially equal since the calculational methodologies are consistent. The predicted core K_{eff} is calculated by a 3D core simulation code as a function of cycle exposure. This calculation is performed for projected or anticipated reactor operating states/conditions throughout the cycle and is usually done prior to cycle operation. The monitored core K_{eff} is the K_{eff} as calculated by the core monitoring system for actual plant conditions.

Since the comparisons are easily done, frequent checks are not an imposition on normal operation. A 1 percent deviation in reactivity from that of the predicted is larger than expected for normal operation and, therefore, should be thoroughly evaluated. A deviation as large as 1 percent would not exceed the design conditions of the reactor.

3/4.1.3 CONTROL RODS

The specification of this section ensure that (1) the minimum SHUTDOWN MARGIN is maintained, (2) the control rod insertion times are consistent with those used in the safety analyses, and (3) limit the potential effects of the rod drop accident. The ACTION statements permit variations from the basic requirements but at the same time impose more restrictive criteria for continued operation. A limitation on inoperable rods is set such that the resultant effect on total rod worth and scram shape will be kept to a minimum. The requirements for the various scram time measurements ensure that any indication of systematic problems with rod drives will be investigated on a timely basis.

Damage within the control rod drive mechanism could be a generic problem, therefore with a control rod immovable because of excessive friction or mechanical interference, operation of the reactor is limited to a time period which is reasonable to determine the cause of the inoperability and at the same time prevent operation with a large number of inoperable control rods.

Control rods that are inoperable for other reasons are permitted to be taken out of service provided that those in the nonfully inserted position are consistent with the SHUTDOWN MARGIN requirements.

The number of control rods permitted to be inoperable could be more than the eight allowed by the specification, but the occurrence of eight inoperable rods could be indicative of a generic problem and the reactor must be shutdown for investigation and resolution of the problem.

The control rod system is designed to bring the reactor subcritical at a rate fast enough to prevent the MCPR from becoming less than the fuel cladding safety limit during the core wide transient analyzed in XN-NF-85-143. This analysis shows that the negative reactivity rates resulting from the scram with the average response of all the drives as given in the specifications, provide the required protection and MCPR remains greater than the fuel cladding safety limit. The occurrence of scram times longer then those specified should be viewed as an indication of a systemic problem with the rod drives and therefore the surveillance interval is reduced in order to prevent operation of the reactor for long periods of time with a potentially serious problem.

The scram discharge volume is required to be OPERABLE so that it will be available when needed to accept discharge water from the control rods during a reactor scram and will isolate the reactor coolant system from the containment when required.

Control rods with inoperable accumulators are declared inoperable and Specification 3.1.3.1 then applies. This prevents a pattern of inoperable accumulators that would result in less reactivity insertion on a scram than has been analyzed even though control rods with inoperable accumulators may still be inserted with normal drive water pressure. Operability of the accumulator ensures that there is a means available to insert the control rods even under the most unfavorable depressurization of the reactor.

CONTROL RODS (Continued)

Control rod coupling integrity is required to ensure compliance with the analysis of the rod drop accident in the FSAR. The overtravel position feature provides the only positive means of determining that a rod is properly coupled and therefore this check must be performed prior to achieving criticality after completing CORE ALTERATIONS that could have affected the control rod coupling integrity. The subsequent check is performed as a backup to the initial demonstration.

In order to ensure that the control rod patterns can be followed and therefore that other parameters are within their limits, the control rod position indication system must be OPERABLE.

The control rod housing support restricts the outward movement of a control rod to less than 3 inches in the event of a housing failure. The amount of rod reactivity which could be added by this small amount of rod withdrawal is less than a normal withdrawal increment and will not contribute to any damage to the primary coolant system. The support is not required when there is no pressure to act as a driving force to rapidly eject a drive housing.

The required surveillance intervals are adequate to determine that the rods are OPERABLE and not so frequent as to cause excessive wear on the system components.

3/4.1.4 CONTROL ROD PROGRAM CONTROLS

Control rod withdrawal and insertion sequences are established to assure that the maximum insequence individual control rod or control rod segments which are withdrawn at any time during the fuel cycle could not be worth enough to result in a peak fuel enthalpy greater than 280 cal/gm in the event of a control rod drop accident. The specified sequences are characterized by homogeneous, scattered patterns of control rod withdrawal. When THERMAL POWER is greater than 20% of RATED THERMAL POWER, there is no possible rod worth which, if dropped at the design rate of the velocity limiter, could result in a peak enthalpy of 280 cal/gm. Thus requiring the RSCS and RWM to be OPERABLE when THERMAL POWER is less than or equal to 20% of RATED THERMAL POWER provides adequate control.

The RSCS and RWM provide automatic supervision to assure that out-of-sequence rods will not be withdrawn or inserted.

Parametric Control Rod Drop Accident analyses have shown that for a wide range of key reactor parameters (which envelope the operating ranges of these parameters) 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 inputs to the parametric analyses to determine the peak fuel rod enthalpy rise. This value is then compared against the 280 cal/gm design limit to demonstrate compliance for each operating cycle. If cycle-specific values of the above parameters are outside the range assumed in the parametric analysis, 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 XN-NF-80-19 Volume 1.

CONTROL RODS PROGRAM CONTROLS (Continued)

The RBM is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power operation. Two channels are provided. Tripping one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage. This system backs up the written sequence used by the operator for withdrawal of control rods.

3/4.1.5 STANDBY LIQUID CONTROL SYSTEM

The standby liquid control system provides a backup capability for bringing the reactor from full power to a cold, Xenon-free shutdown, assuming that none of the withdrawn control rods can be inserted. To meet this objective it is necessary to inject a quantity of boron which produces a concentration of 660 ppm in the reactor core in approximately 90 to 120 minutes. A minimum quantity of 4587 gallons of solution containing a minimum of 5500 pounds of sodium pentaborate is required to meet this shutdown requirement. There is an additional allowance of 150 ppm in the reactor core to account for imperfect mixing. The time requirement was selected to override the reactivity insertion rate due to cooldown following the Xenon poison peak and the required minimum pumping rate is 41.2 gpm. The minimum storage volume of the solution is established to allow for the portion below the pump suction that cannot be inserted and the filling of other piping systems connected to the reactor vessel. The temperature requirement on the sodium pentaborate solution is necessary to ensure that the sodium pentaborate remains in solution.

With redundant pumps and explosive injection valves and with a highly reliable control rod scram system, operation of the reactor is permitted to continue for short periods of time with the system inoperable or for longer periods of time with one of the redundant components inoperable.

Surveillance requirements are established on a frequency that assures a high reliability of the system. Once the solution is established, boron concentration will not vary unless more boron or water is added, thus a check on the temperature and volume once each 24 hours assures that the solution is available for use.

Replacement of the explosive charges in the valves at regular intervals will assure that these valves will not fail because of deterioration of the charges.

BASES

The specifications of this section assure that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the 2200°F limit specified in 10 CFR 50.46.

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

The peak cladding temperature (PCT) following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is dependent only secondarily on the rod to rod power distribution within an assembly. For GE fuel, the peak clad temperature is calculated assuming a LHGR for the highest powered rod which is equal to or less than the design LHGR corrected for densification. This LHGR times 1.02 is used in the heatup code along with the exposure dependent steady-state gap conductance and rod-to-rod local peaking factor. The Technical Specification AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) for GE fuel is this LHGR of the highest powered rod divided by its local peaking factor which results in a calculated LOCA PCT much less than 2200°F. The Technical Specification APLHGR for ENC fuel is specified to assure the PCT following a postulated LOCA will not exceed the 2200°F limit. The limiting value for APLHGR is shown in Figures 3.2.1-1, 3.2.1-2, and 3.2.1-3 for two recirculation loop operation. These values shall be multiplied by a factor of 0.84 for single recirculation loop operation. This multiplier is determined from comparison of the limiting analysis between two recirculation loop and single recirculation loop operation.

The calculational procedure used to establish the APLHGR shown on Figures 3.2.1-1, 3.2.1-2, and 3.2.1-3 is based on a loss-of-coolant accident analysis. The analysis was performed using calculational models which are consistent with the requirements of Appendix K to 10 CFR Part 50. These models are described in Reference 1 or XN-NF-80-19, Volumes 2, 2A, 2B and 2C, Rev. 1.

3/4.2.2 APRM SETPOINTS

The flow biased simulated thermal power-upscale scram setting and control rod block functions of the APRM instruments limit plant operations to the region covered by the transient and accident analysis. In addition, the APRM setpoints must be adjusted for both two recirculation loop operation and single recirculation loop operation to ensure that the MCPR does not become less than the fuel cladding safety limit or that > 1% plastic strain does not occur in the degraded situation. The scram settings and rod block settings are adjusted in accordance with the formula in this specification when the combination of THERMAL POWER and MFLPD indicates a higher peaked power distribution to ensure that an LHGR transient would not be increased in the degraded condition.

3/4.2.3 MINIMUM CRITICAL POWER RATIO

The required operating limit MCPRs at steady-state operating conditions as specified in Specification 3.2.3 are derived from the established fuel cladding integrity Safety Limit MCPR and an analysis of abnormal operational transients. For any abnormal operating transient analysis evaluation with the initial condition of the reactor being at the steady-state operating limit, it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient assuming instrument trip setting given in Specification 2.2.

To assure that the fuel cladding integrity Safety Limit is not exceeded during any anticipated abnormal operational transient, the most limiting transients have been analyzed to determine which result in the largest reduction in CRITICAL POWER RATIO (CPR). The type of transients evaluated were loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest delta MCPR. When added to the Safety Limit MCPR, the required minimum operating limit MCPR of Specification 3.2.3 is obtained and presented in Table 3.2.3-1.

The evaluation of a given transient begins with the system initial parameters shown in XN-NF-85-143 that are input to a ENC-core dynamic behavior transient computer program. The outputs of this program along with the initial MCPR form the input for further analyses of the thermally limiting bundle. The codes and methodology to evaluate pressurization and nonpressurization events are described in XN-NF-79-71. The principal result of this evaluation is the reduction in MCPR caused by the transient.

The purpose of the $MCPR_f$ of Figure 3.2.3-1 is to define operating limits at other than rated core flow conditions. At less than 100% of rated flow the required MCPR is the maximum of the rated flow MCPR determined from Table 3.2.3-1 and the reduced flow MCPR determined from Figure 3.2.3-1, $MCPR_f$ assures that the Safety Limit MCPR will not be violated. $MCPR_f$ is only calculated for the manual flow control mode. Automatic flow control operation is not permitted.

MINIMUM CRITICAL POWER RATIO (Continued)

At THERMAL POWER levels less than or equal to 25% of RATED THERMAL POWER, 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 indicates that the resulting MCPR value is in excess of requirements by a considerable margin. During initial start-up testing of the plant, a MCPR evaluation will be made at 25% of RATED THERMAL POWER level with minimum recirculation pump speed. The MCPR margin will thus be demonstrated such that future MCPR evaluation below this power level will be shown to be unnecessary. The daily requirement for calculating MCPR when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR when a limiting control rod pattern is approached ensures that MCPR will be known following a change in THERMAL POWER or power shape, regardless of magnitude, that could place operation at a thermal limit.

3/4.2.4 LINEAR HEAT GENERATION RATE

This specification assures that the Linear Heat Generation Rate (LHGR) in any rod is less than the design linear heat generation even if fuel pellet densification is postulated.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 28 TO FACILITY OPERATING LICENSE NO. NPF-21

WASHINGTON PUBLIC POWER SUPPLY SYSTEM

WPPSS NUCLEAR PROJECT NO. 2

DOCKET NO. 50-397

1.0 INTRODUCTION

By letter dated February 26, 1986, from G. Sorensen, Washington Public Power Supply System, to Director of Nuclear Reactor Regulation (Reference 1), as supplemented by letters dated April 7, 24, 25, 30 and May 22, 1986, Technical Specification changes were proposed for the operation of Nuclear Plant No. 2 (WNP-2) for Cycle 2 (N2C2) with a reload using Exxon manufactured fuel assemblies and Exxon analyses and methodologies. Enclosed in the February 26 letter were requested Technical Specification changes and a number of reports (References 2-5) discussing the reload and analyses done to support and justify the second cycle operation with General Electric (GE) and Exxon fuel and the Technical Specification changes. A subsequent letter (Reference 6) was submitted, providing a supplemental report (Reference 7) describing changes to the fuel loading from that assumed in some of the initial analyses and providing the changes to those analysis results. There were also changes to some of the proposed Technical Specifications in the later letters. On May 22, 1986, the licensee added Figures 3.2.1-4, 3.2.1-5, and 3.2.1-6 to the Technical specifications. These figures replace the licensee's original proposal which would have used existing figures and required operations to adjust those curves by a predetermined multiplier. The staff finds this change to be more practical and find it acceptable. Cycle 2 will be the first use of Exxon fuel and analysis in this reactor. However, similar reloads with Exxon fuel have been done for Dresden 2 and 3, and more recently for Cycles 2 and 3 for Susquehanna 1; and these reloads and the associated Exxon methodologies were extensively reviewed and approved (see for example Reference 8). These methodologies are generally applicable and were used for the most part for N2C2 analyses.

Beyond the use of Exxon-provided reload fuel, there is little that is different about N2C2 from the first cycle, and the proposed Technical Specification changes are primarily related to the use of Exxon fuel and accompanying analyses and methodology, terminology or related operational approaches. In addition to the standard reload changes, the following variations will also be included in N2C2: (a) There will be two Exxon Lead Test (Fuel) Assemblies (LTA) as part of the reload fuel; (b) There will be a new recirculation pump impeller and other parts originally manufactured for the Black Fox reactor to replace defective parts found during Cycle 1 operation; (c) It is proposed that N2C2 be allowed to operate at a condition of up to 106 percent flow and 100 percent power, and analyses and

Technical Specification additions for this operation have been presented; (d) "Normal" Minimum Critical Power Ratio (MCPR) Technical Specification limits, and the transient analyses to determine them, have been based on measured control rod scram insertion times. The reload, its analyses and the above variations will be discussed in the following evaluation.

2.0 EVALUATION

2.1 Reload Description

The N2C2 reload will retain 636 General Electric (GE) fuel assemblies from the first cycle and will add 128 Exxon manufactured XN-1 8X8, 2.72 percent average, 2.89 percent peak radial average U235 enriched fuel assemblies. As noted above, the XN-1 fuel assemblies are similar to those used in the Susquehanna 1 second cycle (S1C2) reload. The loading pattern will be a conventional scatter pattern with low reactivity fuel on the periphery. Two of the 128 assemblies will be the LTA.

This reload of 128 assemblies is discussed in the supplemental report, Reference 7. The original submittal, References 2 and 3, indicated that originally the reload was planned for 196 assemblies, but was later revised to 132, and finally to 128, largely as a result of failure of a recirculation pump in first cycle and consequent revised power load history. Because of these changes, the transient analyses in the original submittal were mostly based on 196 new assemblies, but the results were revised in the supplement. The nuclear design was based on 132 assemblies.

2.2 Fuel Design

The Exxon XN-1 fuel assembly used for N2C2 is essentially the same as that used for the S1C2 reload. There are slight differences in the fuel enrichment and gadolinium placement patterns, but the significant mechanical and thermal-hydraulic design elements are the same and power distributions are similar. The methodologies used for the fuel design and analysis are the same as those developed and approved during the S1C2 reload review and then approved for the Susquehanna 1 Cycle 3 (S1C3) reload. The design and analyses of the XN-1 fuel assembly as used in N2C2 are thus acceptable.

For N2C2 the Technical Specifications will provide for a Linear Heat Generation Rate (LHGR) specification as a function of fuel burnup for the Exxon fuel. A similar specification was accepted for S1C3 as a result of discussions between the NRC staff and Exxon on the need for a LHGR specification. The specification is based on the approved fuel design methodology as discussed in the S1C3 review (Reference 9) and is acceptable.

The mechanical response of Exxon fuel assemblies to design Seismic-LOCA events is essentially the same as for GE assemblies. Similar to the S1C2 and S1C3 reloads, the channel boxes were manufactured for the assemblies to GE design criteria and dimensions, and as in those reviews, the analyses indicating that the design limits are not exceeded are acceptable.

Two of the 128 XN-1 assemblies will be Lead Test Assemblies. The nuclear, thermal-hydraulic and general mechanical design of these LTA will be the same as the standard assemblies. They will differ only in that 8 (non limiting) fuel pins in each of the assemblies will have fuel pellets and clad variations exploring properties such as grain size, fabrication process and clad heat-treat. The safety related fuel assembly parameters are not affected and the introduction of these LTA is acceptable.

2.3 Nuclear Design

The nuclear design for N2C2 has been performed with Exxon methodologies previously reviewed and approved, and which were listed in the review for the S1C2 reload (Reference 8). The nuclear design for N2C2 was done for the core with 132 new assemblies and was described in the original submittal (Reference 2). The new loading will be 128 assemblies. Exxon has examined the small differences between the 132 and 128 new assembly cores and determined that the original analyses are applicable to the 128 assembly core. Our review indicates this to be a reasonable conclusion and it is acceptable.

The fuel loading pattern is a normal type of scattered configuration. The beginning of cycle shutdown margin is 3.1 percent delta k and at minimum conditions is 1.7 percent delta k, well in excess of the required 0.38 percent delta k. The Standby Liquid Control System also fully meets shutdown requirements. These and other N2C2 nuclear design parameters are consistent with the 128 assembly loading and have been obtained with previously approved methods and fall within expected ranges. Thus the nuclear design is acceptable.

WNP-2 will use the Exxon POWERPLEX core monitoring system to monitor reactor parameters. We have not specifically reviewed details of this system (nor have we in the past reviewed details of the GE process computer monitoring system), but we have reviewed the principal methodologies involved in the system and consider them to be appropriate and acceptable. The system has been in use in Susquehanna and has provided suitable monitoring and predictive results.

2.4 Thermal Hydraulic Design

The Exxon thermal-hydraulic methodology and criteria used for the N2C2 design and analysis is the same as that used and approved in the S1C2 and S1C3 reloads. As was the case for the Susquehanna reloads, statistical aspects of the methodology for which the reviews are incomplete were not needed since bounding transient analyses were used. The previous reviews concluded that hydraulic compatibility between GE and Exxon fuel is satisfactory and the calculation of core bypass flow and the Safety Limit MCPR are acceptable. This is also the case for N2C2. The Safety Limit MCPR continues to be 1.06 for two recirculation loop operation (the same value as for the first cycle GE methodology) as it is for Susquehanna, and this is acceptable. The Operating Limit MCPR is discussed later.

WNP-2 already has Technical Specifications from Cycle 1 allowing and controlling one recirculation loop operation, including changes required on limits for Maximum Average Planar Linear Heat Generation Rate (MPLHGR), Average Power Range Monitor (APRM) settings, and Safety Limit MCPR. Since the Exxon fuel is hydraulically compatible with the GE fuel the previous analyses are also applicable to the Exxon XN-1 fuel loading.

Similar to the approval for the Susquehanna one loop operation review (Reference 10), the above first cycle one loop limit changes are also acceptable for this Exxon reload. WNP-2 also had Technical Specifications approved during the first cycle for Thermal-Hydraulic Stability surveillance and the subsequent suppression of possible oscillations. These specifications are also applicable to N2C2 and thus further review of stability is not necessary for this cycle, although Exxon calculations indicate stability is equivalent to or better than first cycle.

WNP-2 has requested to be allowed to operate at 106 percent flow at 100 percent power for the second cycle. The request is based primarily on GE mechanical and system analyses for Cycle 1 operation presented in Reference 11. This has been augmented by Exxon mechanical analyses for the XN-1 fuel as well as parameter and transient analyses at this statepoint to determine limiting conditions of operation in the extended flow region. The GE analyses examined the effects of increased pressure differential (due to increased flow) on reactor internals, fuel channels and fuel bundles and the effect on flow-induced vibration response of internals, in order to show that design limits would not be exceeded. They also examined the effects of increased flow on containment LOCA response, including LOCA related pool conditions. Standard methodologies were used throughout the analyses and the results were satisfactory. The Exxon review indicated that because of the similarity of the Exxon and GE fuel, the analyses are applicable to the reload core. Since all relevant areas have been covered in these analyses and acceptable methodologies have been used, the extended flow region is acceptable. The relevant MCPR limits are discussed later.

2.5 Transient and Accident Analyses

The originally submitted transient analyses (References 2 and 4) were calculated for a 196 XN-1 assembly core. The relevant transients were reanalyzed for a 132 (equivalent to 128) XN-1 assembly core and reported in Reference 7.

The Exxon transient methodology is the same as that used and approved for the Susquehanna reloads (listed and discussed in Reference 8). The only aspects of the methodology review not yet completed involve statistical analyses which were not used for N2C2 since bounding parameters were used in the calculations.

Exxon examined the standard transient events and the submittal presented results for the more limiting events. The most limiting corewide transient, setting some of the Operating Limit MCPR values, is the Load Rejection Without Bypass (LRWB). The other event setting MCPR limits is the Control Rod Withdrawal Error. These events were analyzed at both 100 percent and 106 percent flow conditions, with both "normal" (defined below) and standard Technical Specification required scram times, and with Recirculation Pump

Trip (RPT) operable and inoperable. These various analyses were used to determine the Technical Specification MCPR operating limits. The selected CRWE analysis used a Rod Block setting of 106 percent in determining the Technical Specification limit and, for "normal" scram times and RPT operable, is the limiting event. These analyses were done with approved methodologies and the results are acceptable.

"Normal" scram time is based on actual rod speeds determined by measurements at WNP-2 and is given in the Technical Specifications. In the event that surveillance indicates that these times are exceeded, the Technical Specifications MCPR limits revert to those determined using standard Technical Specification scram times. Scram time surveillance specifications are the usual requirements of the Standard Technical Specifications. "Normal" scram insertion time is determined via the slowest measured average insertion time (to specified notches) for each group of 4 rods arranged in a 2X2 array. Our review has concluded that this is a reasonable use of plant measured scram time (comparable to GE option B) and is acceptable.

Reduced flow operation and the Recirculation Flow Run-Up event was analyzed by Exxon for N2C2 for manual flow control (automatic control not allowed). This analysis was discussed in Reference 8 for S1C2. In the Exxon methodology this provides a Technical Specification limit for MCPR as a function of core flow. The operating limit MCPR is then the maximum of this curve and the full flow MCPR limit. The analysis has been done with previously approved methods and the results are acceptable.

Compliance with overpressurization criteria was demonstrated by analysis of the Main Steam Isolation Valve (MSIV) closure with MSIV position switch failure. Six safety-relief valves were assumed out of service. Maximum pressure was 105 percent of vessel design pressure, well under the 110 percent criterion. The calculation was done with approved methodology and the results are acceptable.

Because of the difficulties with the loop B recirculation pump during the first cycle, WNP-2 has replaced the pump impeller with one of similar design intended for the Black Fox reactor. The new impeller is slightly smaller than the original, the result of being trimmed to meet specified flow requirements. The results are slightly reduced full flow capacity and inertial moment. The transients and accident analyses have been examined for the effects of this replacement (Reference 7). For the transient analyses the conclusion is that either the event is not impacted

by the impeller, e.g., the CRWE, or the event is slightly improved because of lower inertia, leading to faster flow coastdown and thus increased core voiding and smaller delta CPR. Thus the analysis concludes that transients are not adversely affected. This is acceptable. LOCA effects will be discussed next.

The LOCA analysis for N2C2 was performed with essentially the same Exxon methodology previously approved and used for the S1C2 and S1C3 reloads (References 8 and 9). This analysis is used to provide MAPLHGR limits as a function of burnup for the XN-1 fuel for N2C2. The basic analyses were performed with the generically approved methodologies used for Susquehanna analysis. Exxon also performed a (BWR-5) break spectrum analysis (Reference 12) parallel (and using similar methodology) to that for the BWR-4 break spectrum analysis used for the Susquehanna calculations (Reference 8). This analysis determined the limiting break to be a 3.04ft^2 split break in the recirculation suction piping, which is consistent with the previous GE analysis. Analyses were performed at 106 percent flow to include the extended flow region. The analysis was reexamined and partly redone to determine the effects of the Black Fox impeller (Reference 7). The effect on the break spectrum and impeller placement in either the broken or unbroken loop were investigated. The overall analysis indicated that the peak clad temperature change with the new impeller is small, less than 20°F and the originally calculated MAPLHGR values remain valid. These LOCA analyses have covered an acceptable range of conditions, have been performed with approved methodology and the resulting Technical Specification MAPLHGR values for the XN-1 fuel are acceptable (including operation at 106 percent flow).

The rod drop accident was analyzed with Exxon methodology. The resulting maximum fuel enthalpy of 98 cal/gm is well below the limit of 280 cal/gm. The analysis and result are acceptable.

Our review of the transient and accident analyses done for N2C2 indicates that appropriate methodology and input have been used and the results provide a suitable basis for N2C2 Technical Specifications.

2.6 Technical Specification Changes

The following WNP-2 Technical Specifications and Bases changes have been requested to accommodate the change to Exxon fuel and methodology, operation at 106 percent flow and use of "normal" scram times. For the most part these changes are the same as those approved for S1C2 (or S1C3) on changing to Exxon methodology. The only significant differences relate to scram time definitions and the use of "normal" scram time in the WNP-2 specifications.

(1) Definitions are added for:

(a) Average Bundle Exposure; this is necessary to match the parameter used in Exxon methodology for MAPLHGR and is acceptable,

(b) Critical Power Ratio; this changes to the Exxon XN-3 correlation and is acceptable.

(2) 3/4.12: The change to the definition of reactivity anomaly from control rod density to a monitored k_{eff} anomaly, reflects the use of a more direct parameter. POWERPLEX, which maintains a consistent methodology between active determination and prediction, can monitor k_{eff} directly. The change is acceptable.

(3) 3/4.1.3.3 and 3.1.3.4: The scram time average for all rods is removed (in Reference 7) since it is not used in the transient analysis. The definition for the average scram time for 2X2 arrays of rods is changed (conservatively) to include all four rods rather than just the three fastest rods. This specification is used in the analyses. These changes are acceptable.

(4) 3.2.1, and Figures 3.2.1-1 through 3: This is a change to the use of the Exxon definition of Average Bundle Exposure for Exxon fuel and the transfer to metric units for GE fuel burnup. A MAPLHGR curve is added for the XN-1 fuel and the curve for unused low enrichment GE fuel is removed. These are acceptable changes.

(5) 3/4.2.3 plus Table 3.2.3-1 plus Figure 3.2.3-1: This change removes the elements of the GE methodology for determining MCPR limits, including the K_f function, and replaces them with the results of the Exxon methodology and analyses for N2C2. The new MCPR limits are principally single value functions of (1) GE or Exxon fuel, (2) Scram time, (3) RPT operability and (4) Core flow. MCPR is limited, for reduced flow operation, as given in the figure. As previously discussed these values are the results of Exxon's calculations of transients and are primarily controlled by the RWE and LRWB. The values to be used for Table 3.2.3-1 are not those of the original submittal, but those of Reference 7 from analyses using the revised loading parameters. "Normal" scram time is defined in this specification, including the time to standard notches and surveillance is referenced to the existing surveillance specification of 4.1.3.1. These changes are acceptable.

(6) 3.2.4 and Figure 3.2.4-1: A LHGR for the Exxon XN-1 fuel is added to this specification. As was previously discussed, this type of specification and figure giving LHGR as a function of burnup was added to the Susquehanna 1 specification as a result of staff discussions with Exxon. This addition to the WNP-2 specification is also acceptable.

(7) 3.3.4.2: This change reflects the fact that in 3/4.2.3 MCPR limits are available from calculations with RPT not in operation. Thus operation can continue if these MCPR limits are met. This is acceptable.

(8) There are also minor changes to the index and to the Bases related to the above specification changes. These are changes to components of Bases 2.0, 2.1, 3/4.1 and 3/4.2. In each case these add to, subtract from or change the Bases in order to refer to Exxon fuel, terminology, methodology and references or remove unneeded GE methodology. These changes are similar to those approved for S1C2. These are acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

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

4.0 CONCLUSION

The Commission made a proposed determination that the amendment involves no significant hazards consideration which was published in the Federal Register (51 FR 15416) on April 23, 1986, and consulted with the state of Washington. No public comments were received, and the state of Washington did not have any comments.

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

Principal Contributor: Howard Richings, NRR

Dated: May 23, 1986

REFERENCES

1. Letter from G. S. Sorensen, Washington Public Power Supply System (WPPSS), to E. Adensam, NRC dated February 26, 1986, "Nuclear Plant No. 2 - Reload License Amendment (Cycle 2)"

2. WPPSS - EANF - 101, February 1986, "WNP-2 Cycle 2 Reload Summary Report" and attachment "Technical Specification Changes"

3. XN-NF-86-01 Rev. 1, dated February 1986, "WNP-2 Cycle 2 Reload Analysis"

4. XN-NF-85-143 Rev. 1, dated February 1986 "WNP-2 Cycle 2 Plant Transient Analysis"

5. XN-NF-85-139, dated December 1985, "WNP-2 LOCA-ECCS analyses, MAPLHGR Results"

6. Letter from G. Sorensen, WPPSS, to E. Adensam, NRC, dated April 24, 1986, "Supplement"

7. WPPSS-EANF-101, Supplement, April 1986, "WNP-2 Cycle 2 Reload Summary Report"

8. Letter from W. Butler, NRC to N. W. Curtis, Pennsylvania Power & Light (PP&L), dated May 22, 1985

9. Letter from E. Adensam, NRC, to H. W. Keiser, PP&L, dated April 11, 1986 "Amendment No. 57 - Susquehanna Steam Electric Station, Unit 1"

10. Letter from E. Adensam, NRC to H. W. Keiser, PP&L, dated April 11, 1986, "Amendment Nos. 56 and 26 to - Susquehanna Steam Electric Station Units 1 and 2"

11. Letter from G. Sorensen, WPPSS, to E. Adensam, NRC, dated April 30, 1986, enclosing the report NEDC-31107, March 1986, "Safety Review of WPPSS Nuclear Project No. 2 at Core Flow Conditions above Rated Flow Throughout Cycle 1 and Final Feedwater Temperature Reduction"

12. XN-NF-85-138 (P), dated December 1985, "LOCA Break Spectrum Analysis for a BWR-5" Transmitted by letter dated April 25, 1986

AMENDMENT NO. 28 TO FACILITY OPERATING LICENSE NO. NPF-21
WPPSS NUCLEAR POJECT NO. 2

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