REVISED TECHNICAL SPECIFICATION PAGES
FUEL CYCLE 14 OPERATION
COOPER NUCLEAR STATION
NRC DOCKET NO. 50-298, DPR-46

Revised Pages:	1	17	211a	212e (deleted)
	5	99	211b	212f (deleted)
	6	102	212	212g (deleted)
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Nebraska Public Power District will be loading a new fuel type during the next refueling outage at Cooper Nuclear Station (CNS). This new fuel type is of the GE8X8NB fuel design which was previously reviewed by the NRC and found acceptable for use in Amendment No. 18 to the General Electric Topical Report NEDE-24011-P-A "General Electric Standard Application for Reactor Fuel" (GESTAR). Use of this new fuel design necessitates changes to the CNS Technical Specifications to incorporate new Minimum Critical Power Ratio (MCPR) limits, new Linear Heat Generation Rate (LHGR) limits, and new Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limits with appropriate reduction factors for single loop operation. Additional Technical Specification changes are needed to allow for use of enhanced analytical methodologies performed in accordance with GESTAR for the new fuel design. Specifically, these enhanced methodologies are:

1. Use of the GEXL-PLUS boiling transition thermal limits correlation in place of the GEXL correlation.

2. Use of the GEMINI/ODYN Transient analysis methods for the evaluation of pressurization events. This will replace use of the GENESIS/ODYN models.

The application of these methodologies have been previously reviewed by the NRC and found acceptable as documented in GESTAR. Acceptance of the GEXL-PLUS correlation is documented in Amendment 15 to GESTAR, while acceptance of the GEMINI Transient methods is contained in Amendment 11. In particular, use of the GEXL-PLUS correlation affects the CNS Technical Specification by changing the values of the K_f factor which adjusts the MCPR operating limit while operating at less than rated core flows. The GEMINI Transient models, among other things, make use of different initial power level assumptions than those used by the GENESIS method. Initial power levels different than those described in the CNS Technical Specification Definitions were used, thus necessitating a change.

In addition, this license amendment request includes lowering the safety limit MCPR from 1.07 to 1.06. As shown in the Supplemental Reload Licensing Submittal attached to this request the reload core will consist of four (4) fuel designs; all manufactured by General Electric:

1. Type BP8DRB283 and Type BP8DRB265L, both designated as BP8X8R in GESTAR.

2. 1988 LTAs found acceptable for use in CNS License Amendment No. 118 dated April 1988.

3. Type GE9B-P8DWB302-10GZ-80M-150T which is discussed as GE8X8NB in GESTAR.

4. GE11LTA of which information will be submitted to the NRC under separate correspondence in accordance with CNS Technical Specification 5.2.C.

The 1.06 Safety Limit MCPR value is justified based on the following:

1. Amendment No. 18 to GESTAR states that the safety limit MCPR value for a D-lattice core is 1.06 for the GE8X8NB fuel design. CNS uses D-Type lattice fuel as documented in Table S-1 of the United States Supplement to GESTAR.

2. The GE11LTA has a safety limit MCPR identical to that of the GE8X8NB fuel.

3. The 1988 LTAs are similar to the BP8DRB265L fuel bundle type and both of these types will have experienced two cycles of operation at the commencement of Cycle 14 operation. As indicated in the attached supplemental reload submittal, the BP8X8R/1988 LTA fuel design will have a bundle R-factor of 1.051. Amendment 14 to the GESTAR allows a MCPR safety limit of 1.04 to be applied to D-lattice plant with a core which is operated with second successive reloads of high bundle initial R-factor ($\geq$ 1.04) GE fuel including those of the BP8X8R design. For Cycle 14 operation, CNS will have met the second successive reload of BP8X8R fuel criteria and the 1.04 MCPR safety limit will apply.

Taking into consideration the safety limit MCPR for the various fuel types, a value of 1.06 will bound all the fuel types and is being proposed in this license amendment request.

The license amendment request also includes changing the Technical Specification Section No. 5, "Major Design Features" description of the reactor fuel. Specification 5.2.A currently lists the allowed fuel designs for the core (i.e., 7X7, 8X8, 8X8R, P8X8R and BP8X8R). The District is proposing to revise this specification to delete reference to specific fuel types and replace it with a more generic description of the fuel with the caveat that the fuel design is limited to those approved by the NRC for use in BWRs. Once implemented, this will allow District use of future NRC approved GE fuel designs without the processing and burden of future license amendment requests for both the NRC and the District. The number of fuel assemblies allowed in the core is unchanged by this request. The proposed generic wording for the specification is similar to that previously approved for the River Bend Technical Specifications.

Listing of the Proposed Changes

Nebraska Public Power District requests the following changes be made to the CNS Technical Specifications:

Page No.	Description of Change
1	Revise Definition 1.A.1 of Critical Power Ratio to delete reference to the GEXL correlation and replace it with the term "NRC approved critical power correlation."

Revise Definition 1.A.4 to incorporate the design LHGR values for the GE8X8NB, GE11LTA and 1988 LTA bundles.

Revise Definition 1.D to delete the statement that the design power is the power to which the safety analysis applies. The GEMINI model uses different input power levels for the transient events.

5 Revise Definition 1.Q to delete reference to the Design Power that was redundant to Definition 1.D.

6 Revise Specification 1.1.A to change the safety limit MCPR value to 1.06 for two recirculation loop operation and 1.07 for single-loop operation.

7 Revise Specification 2.1.A.1.a to incorporate the limiting power density values for the GE8X8NB, GE11LTA, and 1988 LTA fuel bundles.

8 Revise Specification 2.1.A.1.d to incorporate the limiting power density values for the GE8X8NB, GE11LTA, and 1988 LTA fuel bundles.

12 (Bases) Revise Reference 1 to reflect the correct title of the document.

17 (Bases)	Delete the phrase "up to 105% of rated steam flow" from the first sentence of the page.
99 (Bases)	Correct a reference to the Updated Safety Analysis Report in the first paragraph on the page.
102 (Bases)	Change the Reference Number from 3 to 2 in the first paragraph of bases for Section C. This corrects a minor editorial error.
104 (Bases)	Revise Reference 1 to reflect the correct title of the document.
	Revise Reference 2 to delete the phrase "Unit 1" from the title of the document.
210	Incorporate the MAPLHGR single loop reduction factors for the GE8X8NB, GE11LTA and the 1988 LTA fuel bundles in Specification 3.11.A.
	Revise Specification 3.11.B to incorporate the LHGR limits for the BP8X8R, 1988 LTA, GE8X8NB and GE11LTA fuel bundles and delete the LHGR equation that contains a maximum power spiking penalty.
211	Revise Figure 3.11-1.1 to contain the MAPLHGR versus exposure curve and data coordinates for the GE8X8NB fuel.

211a	Revise Figure 3.11-1.2 to contain the MAPLHGR versus exposure curve and data coordinates for the GE11LTA fuel.
211b	Renumber the figures to 3.11-1.3 and 3.11-1.4 and revise the title of the upper figure to include the 1988 LTA fuel.
212	Revise Specification 3.11.C to state that K_f is as calculated in Table 3.11.1 instead of being shown in Figure 3.11-3.
212b	Revise Figure 3.11-2a to provide the MCPR curve for the GE8X8NB fuel for Cycle 14 operation.
212c	Revise Figure 3.11-2b to provide the MCPR curve for the GE11LTA fuel for Cycle 14 operation.
212d	Revise Figure 3.11-2c to provide the MCPR curve for the BP8X8R and 1988 LTA fuel for Cycle 14 operation.
212e, f, g	These pages are deleted.
213	Replace Figure 3.11-3 with Table 3.11.1 to supply the K_f flow dependent MCPR multiplier.
214 (Bases)	Revise the Bases for Specification 3.11.B to delete reference to the LHGR power spike penalty and the application to 8X8 fuels.

Revise the Bases for Specification 3.11.A to replace the term "10 CFR Appendix K" to 10CFR50.46 since it contains the Loss-of-Coolant accident criteria and not 10CFR Appendix K.

Add a sentence discussing the APLHGR single loop reduction factors to the Bases for Specification 3.11.A.

214b (Bases) Revise Reference 1 to reflect the correct title of the document.

Revise Reference 2 to delete the phrase "Unit 1" from the title of the document.

Add Reference No. 10 to the Bases of Specification 3.11.

217 Revise Specification 5.2.A to delete reference to specific fuel types and replace it with a more generic fuel description.

Safety Evaluation

A summary of the results of analysis of the Cycle 14 core with respect to the design basis transients is contained in the attached supplemental reload licensing submittal (Document 23A5996). With the proposed Technical Specification limits for operational MCPR, Average Planar Linear Heat Generation Rate and their single loop reduction factors and design linear heat generation rates, the transients and Design Bases Loss of Coolant Accident will meet acceptable criteria.

A review was also performed to verify that the remaining accident analysis contained in the Updated Safety Analysis Report (USAR) remained bounding. A brief description for the various accidents follows:

1. Refueling Accident - Because the GE8X8NB utilizes a large central water hole in the bundle, the total number of failed rods (and the radiological consequences) resulting from a fuel-handling accident is less for a GE8X8NB bundle than for other bundle designs.

2. Main Steam Line Break - The analysis for this accident depends upon the operating thermal hydraulic parameters of the overall reactor and other factors. The primary factors are the rate of flow through the break, which does not change and upon whether the MCPR safety limit is exceeded. For this reload, the MCPR safety limit is not exceeded and therefore existing analysis remains bounding.

3. Control Rod Drop Accident - Cooper Nuclear Station is a group notch plant operating in the Banked Position Withdrawal sequence as documented in License Amendment No. 117 dated February 23, 1988 to the Cooper Nuclear Station Facility Operating License. As such, the Control Rod Drop Accident analysis may be deleted

from the standard GE-BWR reload package. NEDE-24011-P-A-9-US states that the radiological consequences of this accident, even with a full core of GE8X8NB is below the limits specified in 10CFR100.

Evaluation of this Amendment with Respect to 10CFR50.92

This License Amendment request involves four changes that will be separately evaluated under 10CFR50.92. These four changes are designated as follows:

a) Changing the MCPR Safety Limit from 1.07 to 1.06.

b) Use of GE8X8NB and associated analysis methodologies (GEXL-PLUS, GEMINI) previously reviewed and found acceptable by the NRC.

c) Changing specification 5.2.A from denoting specific fuel types to a more generic description of allowed fuel types.

d) Various editorial corrections to the Technical Specification Bases section.

These revisions include:

1) Correcting the title of NEDE 24011-P-A (latest approved version) on proposed pages 12, 104 and 212b.

2) Correcting an Updated Safety Analysis Report section number referred to on proposed page 99.

3) Correcting a reference number in the Bases for Section 4.3.C on proposed page 102.

4) Correcting the title of Reference 2 on proposed page 104 and 214b.

5) Designating 10CFR50.46 as containing the criteria for ECCS performance instead of 10CFR50 Appendix K on page 214.

The enclosed Technical Specification change is judged to involve no significant hazards based on the following:

1. Does the proposed license amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

<u>Evaluation</u>

a. The proposed change would reduce the Minimum Critical Power Ratio (MCPR) safety limit from 1.07 to 1.06. The MCPR safety limit is set to protect the fuel cladding from undergoing boiling transition following any design basis transient. The MCPR safety limit is defined as the critical power ratio 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 for each fuel type using the methodology described in NEDE-24011-P-A "General Electric Standard Application for Reactor Fuel" (GESTAR). The MCPR safety limit for the fuel types in the Reload 13, Cycle 14 core were determined using accepted GESTAR methodologies, and the most conservative value, 1.06, is used as the proposed limit. For the limiting MCPR event there would be no increase in consequences of any design basis event since use of the GESTAR methodologies assures that the criteria of 99.9% of all fuel rods in the core being expected to avoid boiling transition is met.

b. The proposed change would allow use of GE8X8NB fuel type in the core during plant operation. Use of this fuel type was generically found to be acceptable by the NRC in Amendment 18 to GESTAR with documented restrictions. The fuel design has been analyzed using approved methods documented in GESTAR with the results being within accepted limits. Use of the GE8X8NB fuel has been evaluated against current accident analysis results in the USAR for the Refueling Accident, Rod Drop Accident, Main Steam Line Break and Loss of Coolant Accidents. Present results for these accident analyses remain bounding. As discussed in part a) above, the MCPR safety limit was selected to maintain the fuel cladding integrity safety limit. The GE8X8NB fuel response to analyzed transients was also

performed and appropriate operating limit MCPR values are incorporated into Technical Specifications. Use of GE8X8NB fuel will not increase the probability or consequences of an accident previously evaluated.

c. The proposed change will replace wording in Technical Specification 5.2.A regarding specifically allowed fuel assemblies with more generic wording that will permit a fuel type to be used if it is of a design approved by the NRC for use in BWRs. Before NRC approval, fuel designs will have been analyzed and evaluated to approved methodologies and their impact on generic design basis accidents accepted and documented. When this change is implemented, a licensee will still be required to perform a 10CFR50.59 evaluation to determine if an unreviewed safety question exists for the plant-specific use of that fuel type. This two-tier review will ensure the probability and consequences of a previously evaluated accident are not significantly increased. The proposed change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

d. The proposed change will correct several editorial errors located in the Bases section of the plant's Technical Specifications. This will not affect any accident analysis or equipment response in mitigation of an accident. It involves no changes to plant hardware, procedure, or analyses,

but is instead administrative in nature and will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed license amendment create the possibility for a new or different kind of accident from any accident previously evaluated?

Evaluation:

a. The proposed change would reduce the MCPR safety limit from 1.07 to 1.06. It does not allow any new mode or condition of plant operation different from that currently stated in the plant's Updated Safety Analysis Report, nor are plant controls or equipment modified that would change the plant's response to any accident or transient as given in any current analysis. The proposed change does not create the possibility for a new or different kind of accident from any accident previously evaluated.

b. The proposed change will allow use of GE8X8NB fuel type in the core. The fuel type was previously reviewed and found acceptable for use as documented in Amendment No. 18 to GESTAR. No new mode or condition of plant operation will be authorized by this change. The proposed change will not create the possibility for a new or different kind of accident from any accident previously evaluated.

c. The proposed change will replace wording in Technical Specification 5.2.A regarding specifically allowed fuel assemblies with more generic wording that will permit a fuel type to be used if it is of a design approved by the NRC for use in BWRs. The change will not allow any new mode or condition of plant operation different from that currently stated in the plant's Updated Safety Analysis Report. The proposed change will not create the possibility for a new or different kind of accident from any accident previously evaluated.

d. The proposed change will correct several editorial errors located in the Bases section of the plant's Technical Specifications. The change will not allow any new mode of plant operation nor change the function or capability of any plant hardware. The proposed change is administrative in nature and will not create the possibility for a new or different kind of accident from any accident previously evaluated.

3. Does the proposed license amendment involve a significant reduction in a margin of safety?

Evaluation:

a. The MCPR safety limit is set to protect the fuel cladding from undergoing boiling transition following any design basis

transient. Margin is incorporated into the limit to allow for uncertainties in monitoring the core operating state and in calculating the critical power ratio so that 99.9 percent of all rods do not experience boiling transition following any design basis transient. Although the proposed change will reduce the safety limit MCPR from 1.07 to 1.06, because the safety limit MCPR was determined using methodologies described in GESTAR for the fuel types in use for this reload, the margin of safety is maintained. The proposed change does not involve a significant reduction in a margin of safety.

b. The proposed change will allow use of GE8X8NB fuel type in the core. This fuel type and its associated analysis methodologies were reviewed and found acceptable in Amendment 18 to GESTAR. The GE8X8NB fuel for Cooper Nuclear Station was analyzed using these methods to ensure required margins to safety (e.g., fuel cladding integrity safety limit and reactor coolant system integrity) are maintained. The proposed change does not involve a significant reduction in a margin of safety.

c. The proposed change will replace wording in the plant's Technical Specification 5.2.A regarding specifically allowed fuel assemblies with more generic wording that will permit a fuel type to be used if it is of a design approved by the NRC for use in BWRs. A new fuel design's effect on nuclear safety margins will be evaluated by the NRC as part of their review and acceptance of the design. In addition, a licensee will

be required to evaluate, under 10CFR50.59, if an unreviewed safety question exists for the plant-specific use of that fuel type. This two-tier effort will ensure no significant reduction in safety margins will take place in using fuels approved by the NRC for use in BWRs. The proposed change does not involve a significant reduction in a margin of safety.

d. The proposed change will correct several editorial errors located in the Bases section of the plant's Technical Specifications. The change will not allow any change to limits in the allowed operation of the plant or any instrument setpoints, limiting conditions for operation and surveillance requirements. The proposed change is administrative in nature and will not involve a significant reduction in a margin of safety.

1.0 DEFINITIONS

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

A. Thermal Parameters

1. 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 an NRC approved critical power correlation.

2. Maximum Fraction of Limiting Power Density - The Maximum Fraction of Limiting Power Density (MFLPD) is the highest value existing in the core of the Fraction of Limiting Power Density (FLPD).

3. Minimum Critical Power Ratio (MCPR) - The minimum critical power ratio corresponding to the most limiting fuel assembly in the core.

4. Fraction of Limiting Power Density - The ratio of the linear heat generation rate (LHGR) existing at a given location to the design LHGR for that bundle type. Design LHGR's are 13.4 KW/ft for BP8X8R and 1988 LTA bundles and 14.4 KW/ft for GE8X8NB and GE11 LTA bundles.

5. 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.

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. Normal movement of in-core instrumentation is not defined as a core alteration.

C. Cold Condition - Reactor coolant temperature equal to or less than 212°F.

D. Design Power - Design power means a steady-state power level of 2486 thermal megawatts. This is 104.4% of Rated Power (105% of rated steam flow).

E. Engineered Safeguard - An engineered safeguard is a safety system the actions of which are essential to a safety action required to maintain the consequences of postulated accidents within acceptable limits.

E.A Dose Equivalent I-131 - The DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcurie/gram) which alone would produce the same thyroid dose if inhaled by an adult as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The dose equivalent I-131 concentration is calculated by: equiv. I-131 = (I-131) + 0.0096 (I-132) + 0.18 (I-133) + 0.0025 (I-134) + 0.037 (I-135).

E.B Exhaust Ventilation Treatment System - An EXHAUST VENTILATION TREATMENT SYSTEM (EVTS) is a system intended to remove radioiodine or radioactive material in particulate form from gaseous effluent by passing exhaust ventilation air through charcoal absorbers and/or HEPA filters before exhausting the air to the environment. An EVTS is not intended to affect noble gas in gaseous effluent. Engineered Safety Feature (ESF) gaseous treatment systems are not considered to be EVTS. The Standby Gas Treatment System is an ESF and not an EVTS. EVTS are specifically identified in ODAM Figure 3-1.

3. All automatic containment isolation valves are operable or de-activated in the isolated position.

4. All blind flanges and manways are closed.

P.A Purge - Purging - Purge or Purging is the controlled process of discharging air or gas from a confinement to establish temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

P.B Process Control Program - The Process Control Program outlines the solidification of radioactive waste from liquid systems. It does not substitute for station operating procedures, but provides a general description of equipment, controls, and practices to be considered during waste solidification to assure solid wastes.

Q. Rated Power - Rated power refers to operation at a reactor power of 2381 megawatts thermal. This is also termed 100% power and is the maximum power level authorized by the operating license. Rated steam flow, rated coolant flow, rated neutron flux, and rated nuclear system pressure refer to the values of these parameters when the reactor is at rated power.

R. Reactor Power Operation - Reactor power operation is any operation with the mode switch in the "Startup/Hot Standby" or "Run" position with the reactor critical and above 1% rated power.

S. Reactor Vessel Pressure - Unless otherwise indicated, reactor vessel pressures listed in the Technical Specifications are those measured by the reactor vessel steam space detectors.

T. Refueling Outage - Refueling outage is the period of time between the shutdown of the unit prior to a refueling and the startup of the plant after that refueling.

U. Safety Limits - The safety limits are limits within which the reasonable maintenance of the fuel cladding integrity and the reactor coolant system integrity are assured. Violation of such a limit is cause for unit shutdown and review by the Nuclear Regulatory Commission before resumption of unit operation. Operation beyond such a limit may not in itself result in serious consequences but it indicates an operational deficiency subject to regulatory review.

V. Secondary Containment Integrity - Secondary containment integrity means that the reactor building is intact and the following conditions are met:

1. At least one door in each access opening is closed.

2. The standby gas treatment system is operable.

3. All automatic ventilation system isolation valves are operable or secured in the isolated position.

W. Shutdown - The reactor is in a shutdown condition when the mode switch is in the "Shutdown" or "Refuel" position.

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 and the reactor vessel vented.

SAFETY LIMITS

1.1 FUEL CLADDING INTEGRITY

Applicability

The Safety Limits established to preserve the fuel cladding integrity apply to those variables which monitor the fuel thermal behavior.

Objective

The objective of the Safety Limits is to establish limits below which the integrity of the fuel cladding is preserved.

Action

If a Safety Limit is exceeded, the reactor shall be in at least hot shutdown within 2 hours.

Specifications

A. Reactor Pressure ≥800 psia and Core Flow ≥10% of Rated

The existence of a minimum critical power ratio (MCPR) less than 1.06 for two recirculation loop operation (1.07 for single-loop operation) shall constitute violation of the fuel cladding integrity safety.

B. Core Thermal Power Limit (Reactor Pressure ≤800 psia and/or Core Flow <10%)

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

C. Power Transient

To ensure that the Safety Limit established in Specification 1.1.A and 1.1.B is not exceeded, each required scram shall be initiated by its expected scram signal. The Safety Limit shall be assumed to be exceeded when scram is accomplished by a scram signal.

LIMITING SAFETY SYSTEM SETTINGS

2.1 FUEL CLADDING INTEGRITY

Applicability

The Limiting Safety System Settings apply to trip settings of the instruments and devices which are provided to prevent the fuel cladding integrity Safety Limits from being exceeded.

Objective

The objective of the Limiting Safety System Settings is to define the level of the process variables at which automatic protective action is initiated to prevent the fuel cladding integrity Safety Limits from being exceeded.

Specifications

A. Trip Settings

The limiting safety system trip settings shall be as specified below:

1. Neutron Flux Trip Settings

a. APRM Flux Scram Trip Setting (Run Mode)

When the Mode Switch is in the RUN position, the APRM flux scram trip setting shall be:

$S \leq 0.66\ W + 54\% - .66\ \Delta W$

where:

S = Setting in percent of rated thermal power (2381 MWt)

W = Two-loop recirculation flow rate in percent of rated (rated loop recirculation flow rate is that recirculation flow rate which provides 100% coreflow at 100% power)

ΔW = Difference between two-loop and single-loop effective drive flow at the same core flow.

SAFETY LIMITS

1.1 (Cont'd)

D. Cold Shutdown

Whenever the reactor is in the cold shutdown condition with irradiated fuel in the reactor vessel, the water level shall not be less than 18 in. above the top of the normal active fuel zone (top of active fuel is defined in Figure 2.1.1).

LIMITING SAFETY SYSTEM SETTINGS

2.1.A.1 (Cont'd)

$\Delta W = 0$ for two recirculation loop operation.

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

$$S \leq (0.66\ W + 54\% - 0.66\ \Delta W)\ \frac{FRP}{MFLPD}$$

where,

FRP = fraction of rated thermal power (2381 MWt)

MFLPD = maximum fraction of limiting power density where the limiting power density is 13.4 KW/ft for BP8X8R and 1988 LTA fuel, and 14.4 KW/ft for GE8X8NB and GE11 LTA fuel.

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

For no combination of loop recirculation flow rate and core thermal power shall the APRM flux scram trip setting be allowed to exceed 120% of rated thermal power.

b. APRM Flux Scram Trip Setting (Refuel or Start and Hot Standby Mode)

When the reactor mode switch is in the REFUEL or STARTUP position, the APRM scram shall be set at less than or equal to 15% of rated power.

c. IRM

The IRM flux scram setting shall be ≤120/125 of scale.

SAFETY LIMITS	LIMITING SAFETY SYSTEM SETTINGS

2.1.A.1 (Cont'd)

d. APRM Rod Block Trip Setting

The APRM rod block trip setting shall be:

$S_{RB} \leq 0.66\ W + 42\% - .66\ \Delta W$

where:

S_{RB} = Rod block setting in percent of rated thermal power (2381 MWt)

W and ΔW are defined in Specification 2.1.A.1.a.

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

$$S_{RB} \leq (0.66\ W + 42\% - 0.66\ \Delta W)\ \frac{FRP}{MFLPD}$$

where,

FRP = fraction of rated thermal power (2381 MWt)

MFLPD - maximum fraction of limiting power density where the limiting power density is 13.4 KW/ft for BP8X8R and 1988 LTA fuel, and 14.4 KW/ft for GE8X8NB and GE11 LTA fuel.

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

2. Reactor Water Low Level Scram and Isolation Trip Setting (except MSIV)

$\geq$ +12.5 in. on vessel level instruments.

1.1 Bases: (Cont'd)

C. Power Transient

Plant safety analyses have shown that the scrams caused by exceeding any safety setting will assure that the Safety Limit of Specification 1.1A or 1.1B will not be exceeded. Scram times are checked periodically to assure the insertion times are adequate. The thermal power transient resulting when a scram is accomplished other than by the expected scram signal (e.g., scram from neutron flux following closure of the main turbine stop valves) does not necessarily cause fuel damage. However, for this specification a Safety Limit violation will be assumed when a scram is only accomplished by means of a backup feature of the plant design. The concept of not approaching a Safety Limit provided scram signals are operable is supported by the extensive plant safety analysis.

The computer provided with Cooper has a sequence annunciation program which will indicate the sequence in which events such as scram, APRM trip initiation, pressure scram initiation, etc. occur. This program also indicates when the scram setpoint is cleared. This will provide information on how long a scram condition exists and thus provide some measure of the energy added during a transient. Thus, computer information normaly will be available for analyzing scrams; however, if the computer information should not be available for any scram analysis, Specification 1.1.C will be relied on to determine if a Safety Limit has been violated.

D. Reactor Water Level (Shutdown Condition)

During periods when the reactor is shutdown, consideration must also be given to water level requirements due to the effect of decay heat. If reactor water level should drop below the top of the active fuel during this time, the ability to cool the core is reduced. This reduction in core cooling capability could lead to elevated cladding temperatures and clad perforation. The core can be cooled sufficiently should the water level be reduced to two-thirds the core height. Establishment of tHe safety limit at 18 inches above the top of the fuel provides adequate margin.

References for 1.1 Bases

1. "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A-(latest approved revision).

2. "Cooper Nuclear Station Single-Loop Operation," NEDO-24258, May, 1980.

2.1 Bases:

The abnormal operational transients applicable to operation of the CNS Unit have been analyzed throughout the spectrum of planned operating conditions. The analyses were based upon plant operation in accordance with Reference 3. In addition, 2381 MWt is the licensed maximum power level of CNS, and this represents the maximum steady-state power which shall not knowingly be exceeded.

The transient analyses performed each reload are given in Reference 1. Models and model conservatisms are also described in this reference. As discussed in Reference 2, the core wide transient analyses for one recirculation pump operation is conservatively bounded by two-loop operation analyses and the flow-dependent rod block and scram setpoint equations are adjusted for one-pump operation.

A. Trip Settings

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

1. Neutron Flux Trip Settings

a. APRM Flux Scram Trip Setting (Run Mode)

The average power range monitoring (APRM) system, which is calibrated using heat balance data taken during steady state conditions, reads in percent of rated thermal power (2381 MWt). Because fission chambers provide the basic input signals, the APRM system responds directly to average neutron flux. During transients, the instantaneous rate of heat transfer from the fuel (reactor thermal power) is less than the instantaneous neutron flux due to the time constant of the fuel. Therefore, during abnormal operational transients, the thermal power of the fuel will be less than that indicated by the neutron flux at the scram setting. Analyses demonstrate that with a 120% scram trip setting, none of the abnormal operational transients analyzed violate the fuel Safety Limit and there is a substantial margin from fuel damage. Therefore, the use of flow referenced scram trip provides even additional margin.

3.3 and 4.3 BASES

A. Reactivity Limitation

1. The requirements for the control rod drive system have been identified by evaluating the need for reactivity control via control rod movement over the full spectrum of plant conditions and events. As discussed in subsection III.6 of the Updated Safety Analysis Report (USAR) the control rod system design is intended to provide sufficient control of core reactivity that the core could be made subcritical with the strongest rod fully withdrawn. This reactivity characteristic has been a basic assumption in the analysis of plant performance. Compliance with this requirement can be demonstrated conveniently only at the time of initial fuel loading or refueling. Therefore, the demonstration must be such that it will apply to the entire subsequent fuel cycle. The demonstration shall be performed with the reactor core in the cold, xenon-free condition and will show that the reactor is subcritical by at least R + 0.38% $\Delta k/k$ with the analytically determined strongest control rod fully withdrawn.

The value of "R", in units of $\%\Delta k/k$, is the amount by which the core reactivity, in the most reactive condition at any time in the subsequent operating cycle, is calculated to be greater than at the time of the demonstration. "R", therefore, 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 cycle.

The demonstration is performed with a control rod which is calculated to be the strongest rod. In determining this "analytically strongest" rod, it is assumed that every fuel assembly of the same type has identical material properties. In the actual core, however, the control cell material properties vary within allowed manufacturing tolerances, and the strongest rod is determined by a combination of the control cell geometry and local k_∞. Therefore, an additional margin is included in the shutdown margin test to account for the fact that the rod used for the demonstration (the "analytically strongest") is not necessarily the strongest rod in the core. Studies have been made which compare experimental criticals with calculated criticals. These studies have shown that actual criticals can be predicted within a given tolerance band. For gadolinia cores the additional margin required due to control cell material manufacturing tolerances and calculational uncertainties has experimentally been determined to be 0.38% $\Delta k/k$. When this additional margin is demonstrated, it assures that the reactivity control requirement is met.

2. Reactivity margin - inoperable control rods.

Specification 3.3.A.2 requires that a rod be taken out of service if it

5. The Rod Block Monitor (RBM) is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power level operation. Two channels are provided, and one of these may be bypassed from the console for maintenance and/or testing. Tripping of one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage. This system backs up the operator who withdraws control rods according to written sequences. The specified restrictions with one channel out of service conservatively assure that fuel damage will not occur due to rod withdrawal errors when this condition exists.

   A limiting control rod pattern is a pattern which results in the core being on a thermal hydraulic limit (i.e., MCPR equals the operating limit as defined on Figure 3.11, and LHGR = as defined in 1.0.A.4). During use of such patterns, it is judged that testing of the RBM system prior to withdrawal of such rods to assure its operability will assure that improper withdrawal does not occur. It is the responsibility of the Reactor Engineer to identify these limiting patterns and the designated rods either when the patterns are initially established or as they develop due to the occurrence ofinoperable control rods in other than limiting patterns. Other personnel qualified to perform this function may be designated by the Division Manager of Nuclear Operations.

C. Scram Insertion Times

The control rod system is designed to bring the reactor subcritical at a rate fast enough to prevent fuel damage; i.e., to prevent the MCPR from becoming less than the safety limit. The limiting power transient is defined in Reference 2. Analysis of this transient shows that the negative reactivity rates resulting from the scram provide the required protection, and MCPR remains greater than the safety limit.

The surveillance requirement for scram testing of all the control rods after each refueling outage and 10% of the control rods at 16-week intervals is adequate for determining the operability of the control rod system yet is not so frequent as to cause excessive wear on the control rod system components.

The numerical values assigned to the predicted scram performance are based on the analysis of data from other BWR's with control rod drives the same as those on Cooper Nuclear Station.

The occurrence of scram times within the limits, but significantly longer than the average, should be viewed as an indication of a systematic problem with control rod drives.

In the analytical treatment of the transients which are assumed to scram on high neutron flux, 290 milliseconds are allowed between a neutron sensor reaching the scram point and start of motion of the control rods. This is adequate and conservative when compared to the typical time delay of about 210 milliseconds estimated from scram test results. Approximately the first 90 milliseconds of each of these time intervals result from the sensor and circuit delays; at this point, the pilot scram solenoid deenergizes. Approximately 120 milliseconds later,

3.3 and 4.3 BASES: (Cont'd)

G. Scram Discharge Volume

To ensure the Scram Discharge Volume (SDV) does not fill with water, the vent and drain valves shall be verified open at least once every 31 days. This is to preclude establishing a water inventory, which if sufficiently large, could result in slow scram times or only a partial control rod insertion.

The vent and drain valves shut on a scram signal thus providing a contained volume (SDV) capable of receiving the full volume of water discharged by the control rod drives at any reactor vessel pressure. Following a scram the SDV is discharged into the reactor building drain system.

REFERENCES

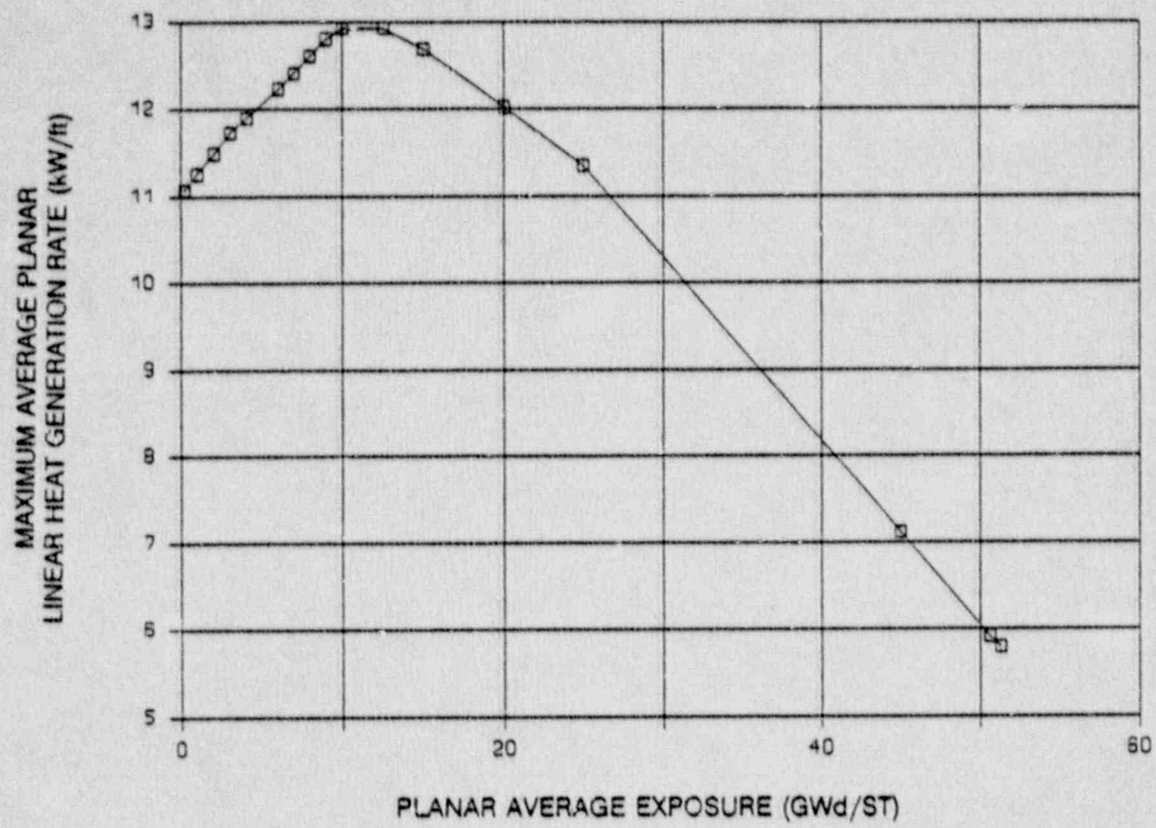
1. "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A-(latest approved revision).

2. "Supplemental Reload Licensing Submittal for Cooper Nuclear Station," (applicable reload document).

3. General Electric Service Information Letter No. 380, Revision 1, dated February 10, 1984.

4. General Electric Service Information Letter No. 316. Reduced Notch Worth Procedure, November, 1979.

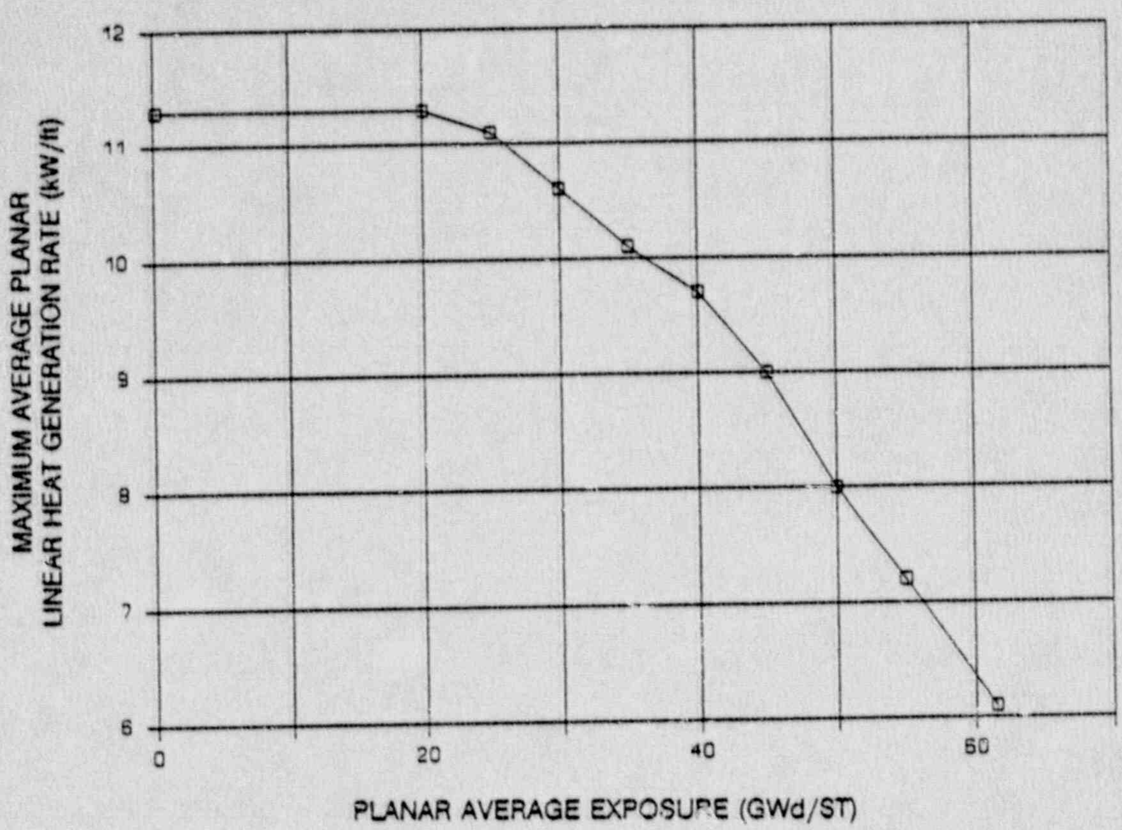
LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.11 FUEL RODS	4.11 FUEL RODS
Applicability The Limiting Conditions for Operation associated with the fuel rods apply to those parameters which monitor the fuel rod operating conditions.	**Applicability** The Surveillance Requirements apply to the parameters which monitor the fuel rod operating conditions.
Objective The Objective of the Limiting Conditions for Operation is to assure the performance of the fuel rods.	**Objective** The Objective of the Surveillance Requirements is to specify the type and frequency of surveillance to be applied to the fuel rods.
Specifications	**Specifications**
A. Average Planar Linear Heat Generation Rate (APLHGR) During steady state power operation, the APLHGR for each type of fuel as a function of average planar exposure shall not exceed the limiting value shown in Figure 3.11-1 for two recirculation loop operation. For single-loop operation, the limits are reduced to 0.77 of the curves' value for the BP8X8R and 1988 LTA fuel and to 0.75 of the curves' value for the GE8X8NB and GE11LTA fuel. If at any time during steady state operation it is determined by normal surveillance that the limiting value for APLHGR is being exceeded action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the APLHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until the prescribed limits are again being met.	A. Average Planar Linear Heat Generation Rate (APLHGR) The APLHGR for each type of fuel as a function of average planar exposure shall be determined daily during reactor operation at ≥25% rated thermal power.
B. Linear Heat Generation Rate (LHGR) During steady state power operation, the linear heat generation rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed 13.4 KW/ft for BP8X8R and 1988 LTA fuel; or 14.4 KW/ft for GE11 LTA and GE8X8NB fuel.	B. Linear Heat Generation Rate (LHGR) The LHGR as a function of core height shall be checked daily during reactor operation at ≥ 25% rated thermal power.



DATA COORDINATES

GWD/ST	kW/ft
0.2	11.07
1.0	11.26
2.0	11.49
3.0	11.73
4.0	11.90
6.0	12.23
7.0	12.41
8.0	12.61
9.0	12.80
10.0	12.93
12.5	12.93
15.0	12.69
20.0	12.02
25.0	11.35
45.0	7.12
50.6	5.92
51.3	5.80

Figure 3.11-1.1 Maximum Average Planar Linear Heat Generation Rate versus Exposure with LPCI Modification and Bypass Holes Plugged, GE8X8NB Fuel



DATA COORDINATES

GWD/ST	kW/ft
0.2	11.3
20.0	11.3
25.0	11.1
30.0	10.6
35.0	10.1
40.0	9.7
45.0	9.0
50.0	8.0
55.0	7.2
61.5	6.1

Figure 3.11-1.2 Maximum Average Planar Linear Heat Generation Rate versus Exposure with LPCI Modification and Bypass Holes Plugged, GE11 LTA Fuel

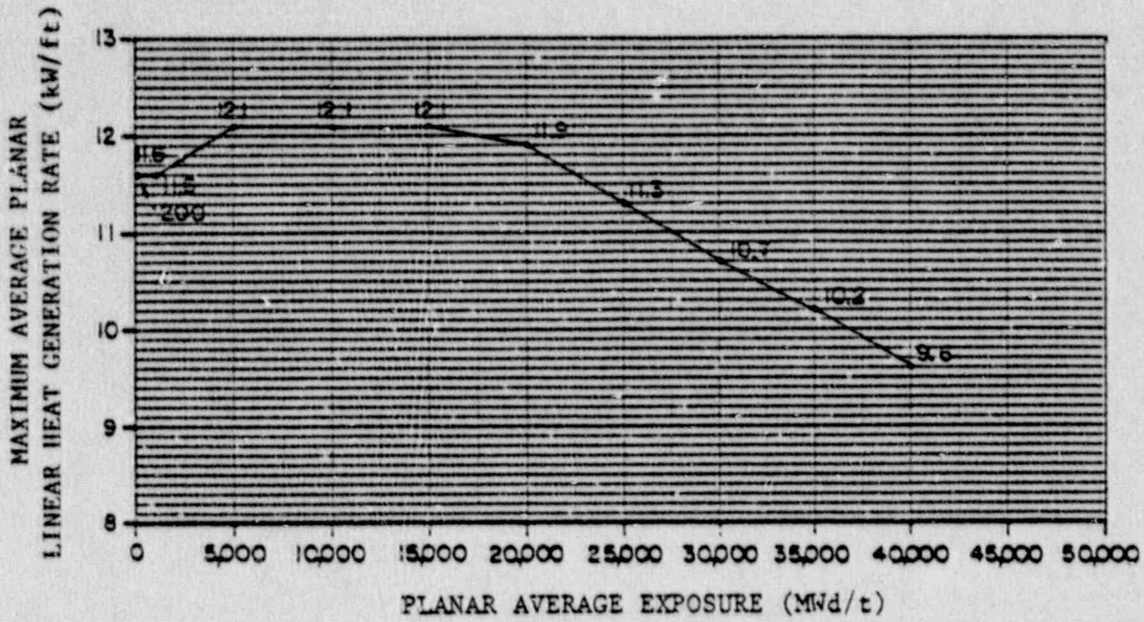


Figure 3.11-1.3 Maximum Average Planar Linear Heat Generation Rate versus Exposure with LPCI Modification and Bypass Flow Holes Plugged, P8DRB265L and BP8DRB265L Fuel and 1988 LTA.

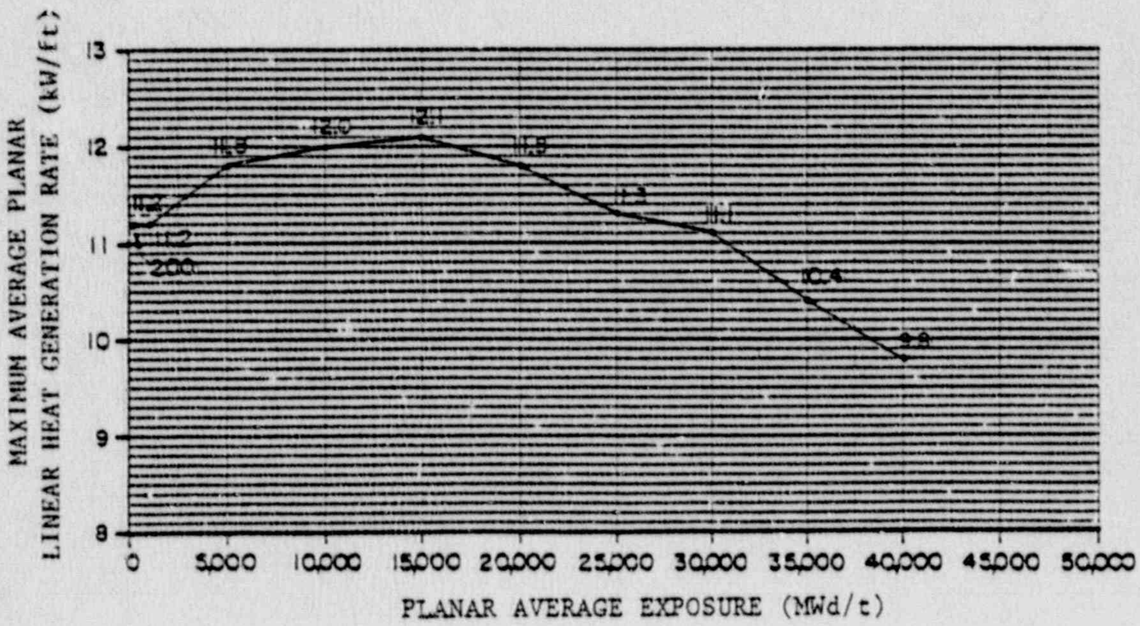


Figure 3.11-1.4 Maximum Average Planar Linear Heat Generation Rate versus Exposure with LPCI Modification and Bypass Flow Holes Plugged, 8DRB283, P8DRB283 and BP8DRB283 Fuel.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
If at any time during steady state operation it is determined by normal surveillance that the limiting value for LHGR is being exceeded action shall then be initiated to restore operation to within the prescribed limits. Surveillance and corresponding action shall continue until the prescribed limits are again being met.	
C. Minimum Critical Power Ratio (MCPR) During steady state power operation the MCPR for each type of fuel at rated power and flow shall not be lower than the limiting value shown in Figure 3.11-2 for two recirculation loop operation. If, at any time during steady state operation it is determined by normal surveillance that the limiting value for MCPR is being exceeded, action shall then be initiated within 15 minutes to restore operation to within the prescribed limits. If the steady state MCPR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until the prescribed limits are again being met. For core flows other than rated the MCPR shall be the operating limit at rated flow times K_f, where K_f is as calculated in Table 3.11.1. For one recirculation loop operation the MCPR limits at rated flow are 0.01 higher than the comparable two-loop values.	C. Minimum Critical Power Ratio (MCPR) MCPR shall be determined daily during reactor power operation at > 25% rated thermel power and following any change in power level or distribution that would cause operation with a limiting control rod pattern as described in the bases for Specification 3.3.B.5.

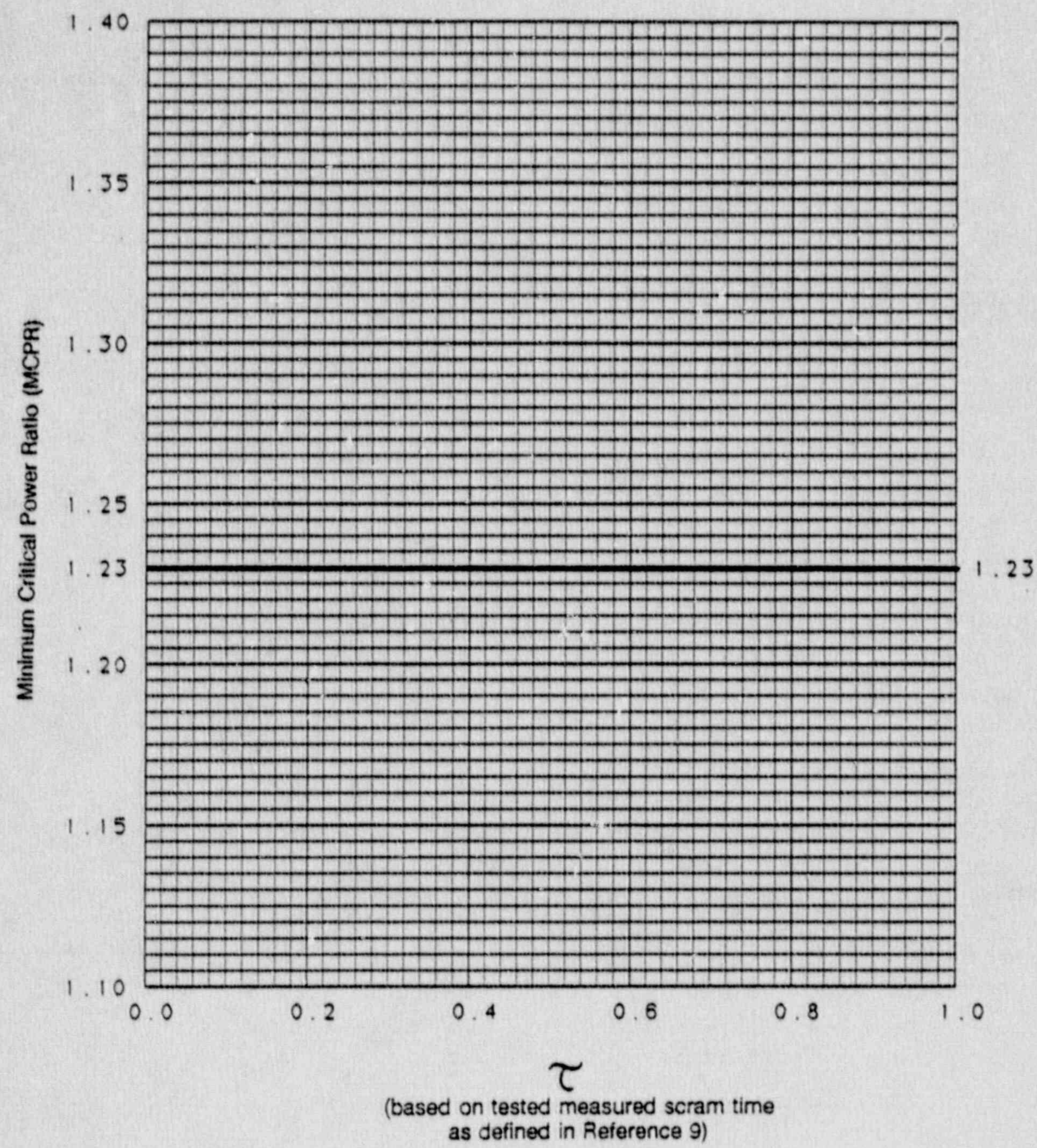


Figure 3.11-2a GE8X8NB Fuel (BOC to EOC)

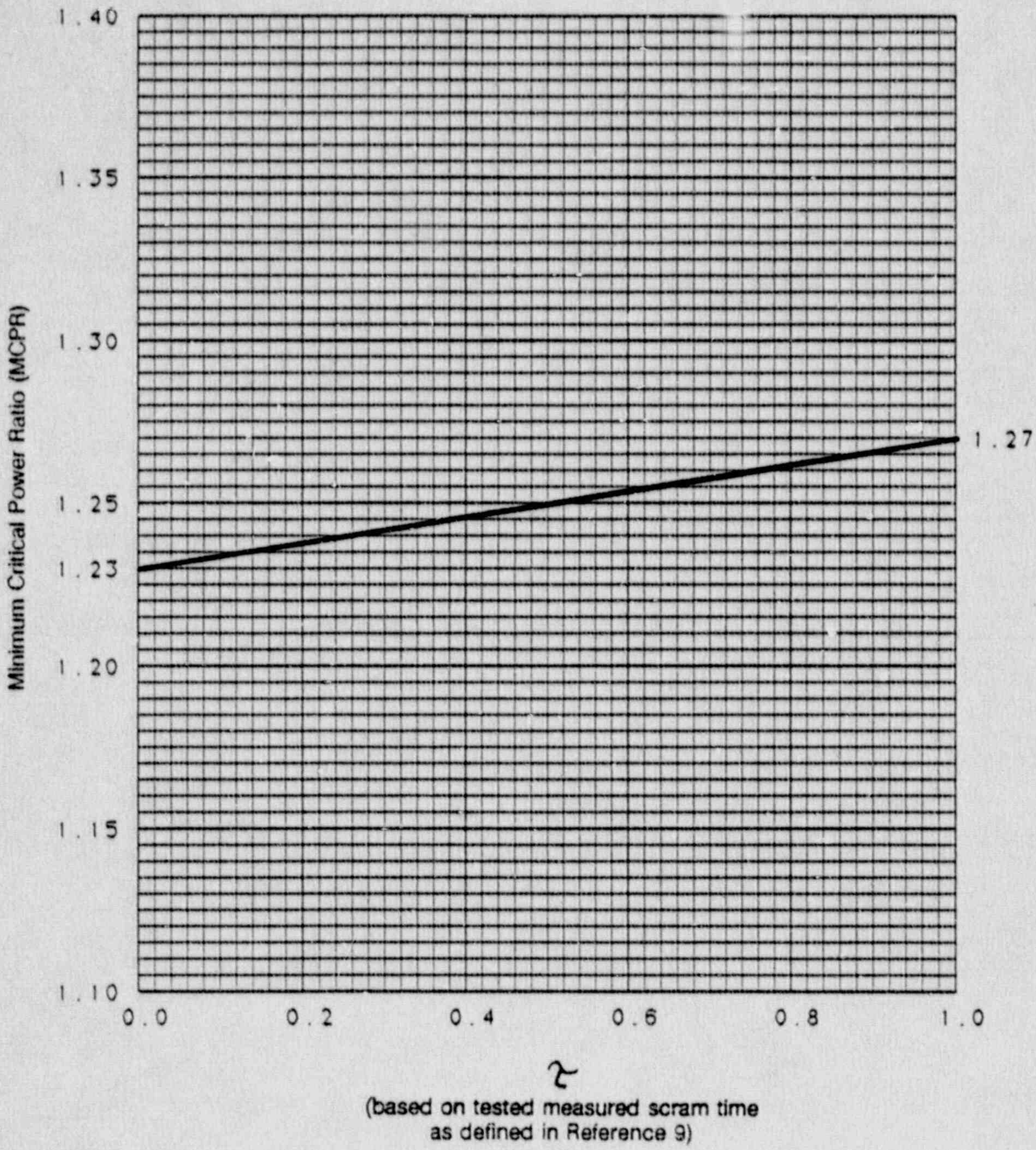


Figure 3.11-2b GE11 LTA Fuel (BOC to EOC)

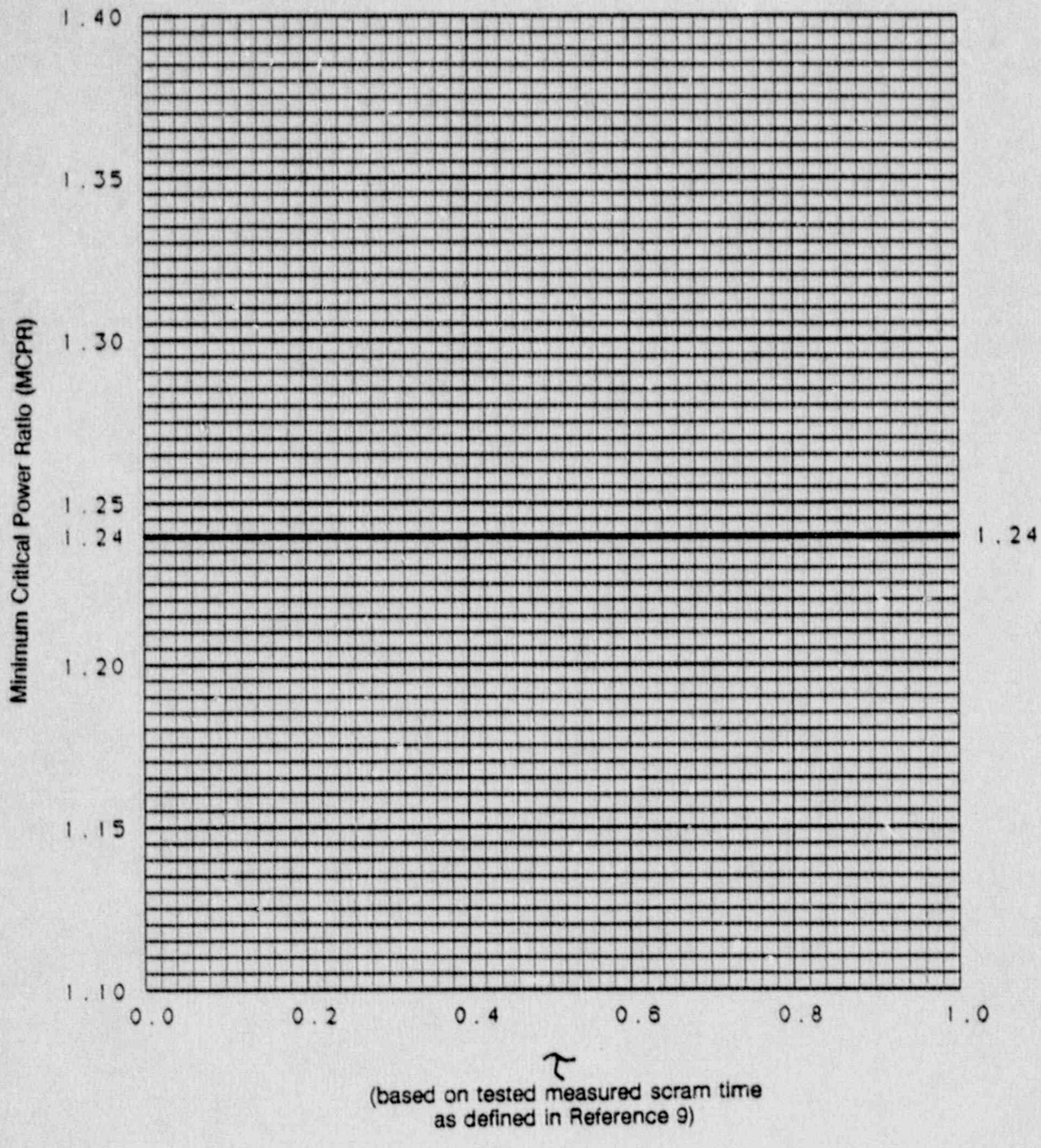


Figure 3.11-2c BP8X8R and 1988 LTA Fuel (BOC to EOC)

Table 3.11.1

BWR/2-4 FLOW DEPENDENT MCPR MULTIPLIER (K_f) WITH GEXL-PLUS LOW FLOW ADJUSTMENT

For $40\% < WT \leq 100\%$, $K_f = MAX\,[1.0,\ A - 0.00441*WT]$
$WT \leq 40\%$, $K_f = [A - 0.00441*WT]*[1.0 + 0.0032*(40-WT)]$

where WT = Percent of Rated Core Flow, and
A = constant which depends on the Flow Control mode and the Scoop Tube Setpoint as noted below.

FLOW CONTROL MODE	SCOOP TUBE SETPOINT	A
MANUAL	102.5%	1.3308
MANUAL	107.0%	1.3528
MANUAL	112.0%	1.3793
MANUAL	117.0%	1.4035
AUTOMATIC	N/A	1.4410

3.11 BASES

A. Average Planar Linear Heat Generation Rate (APLHGR)

This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in 10CFR50.46.

The peak cladding temperature 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 only dependent secondarily on the rod to rod power distribution within an assembly. Since expected local variations in power distribution within a fuel assembly affect the calculated peak clad temperature by less than ± 20°F relative to the peak temperature for a typical fuel design, the limit on the average linear heat generation rate is sufficient to assure that calculated temperatures are within the 10CFR50.46 limit. The limiting value for APLHGR is shown in Figure 3.11-1.

The APLHGR valves are reduced for single loop operation per Reference 10.

B. Linear Heat Generation Rate (LHGR)

This specification assures that the linear heat generation rate in any rod is less than the design linear heat generation if fuel pellet densification is postulated. The LHGR as a function of core height shall be checked daily during reactor operation at ≥ 25% power to determine if fuel burnup, or control rod movement has caused changes in power distribution. For LHGR to be a limiting value below 25% rated thermal power, the MTPF would have to be greater than 10 which is precluded by a considerable margin when employing any permissible control rod pattern. Pellet densification power spiking in GE fuel has been accounted for in the safety analysis presented in References 1 and 2; thus no adjustment to the LHGR limit for densification effects is required.

3.11 <u>Bases</u>: (Cont'd)

The K_f factors as calculated by Table 3.11.1 were developed generically which are applicable to all BWR/2, BWR/3, and BWR/4 reactors. The K_f factors were derived using the flow control line corresponding to rated thermal power at rated core flow as described in Reference 1.

The K_f factors are conservative for Cooper operation because the operating limit MCPR's are greater than the original 1.20 operating limit MCPR used for the generic derivation of K_f.

<u>References for Bases 3.11</u>

1. "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A-(latest approved revision).

2. "Supplemental Reload Licensing Submittal for Cooper Nuclear Station," (applicable reload document).

3-8. Deleted

9. Letter (with attachment), R. H. Buckholz (GE) to P. S. Check (NRC), "Response to NRC Request for Information on ODYN Computer Model," September 5, 1980.

10. "Cooper Nuclear Station Single-Loop Operation," NEDO 24258.

5.0 MAJOR DESIGN FEATURES

5.1 Site Features

The Cooper Nuclear Station site is located in Nemaha County, Nebraska, on the west bank of the Missouri River, at river mile 532.5. This part of the river is referred to by the Corps of Engineers as the Lower Brownville Bend. Site coordinates are approximately 40° 21' north latitude and 95° 38' west longitude. The site consists of 1351 acres of land owned by Nebraska Public Power District. About 205 acres of this property is located in Atchison County, Missouri, opposite the Nebraska portion of the station site. The land area upon which the station is constructed is crossed by the Missouri River on the east and is bounded by privately owned property on the north, south, and west. At the west site boundary, a county road and Burlington Northern Railroad spur pass the site.

The reactor (center line) is located approximately 3600 feet from the nearest property boundary. No part of the present property shall be sold or leased by the applicant which would reduce the minimum distance from the reactor to the nearest site boundary to less than 3600 feet without prior NRC approval.

The protected area is formed by a seven foot chain link fence which surrounds the site buildings.

5.2 Reactor

A. The reactor shall contain 548 fuel assemblies. Each assembly shall consist of a matrix of Zircalloy clad fuel rods with an initial composition of slightly enriched uranium dioxide (UO_2) as fuel material. Fuel assemblies shall be limited to those fuel designs approved by the NRC for use in BWRs.

B. The core shall contain 137 cruciform-shaped control rods. The control material shall be boron carbide powder (B_4C) compacted to approximately 70% theoretical density, except for the Hybrid I control rods which contain approximately 15% hafnium.

C. Lead Test Assembly (LTA) control blades and fuel assemblies of different design than described above may be installed under the provisions of 10CFR50.59 in conjunction with vendor test programs. The LTAs shall have been analyzed using methods previously approved by the NRC. The licensee will provide the NRC with a report describing the LTAs and analyses not less than 30 days prior to startup.

5.3 Reactor Vessel

The reactor vessel shall be as described in Section IV-20 of the SAR. The applicable design shall be as described in this section of the SAR.

5.4 Containment

A. The principal design parameters for the primary containment shall be as given in Table V-2-1 of the SAR. The applicable design shall be as described in Section XII-2.3 of the SAR.

B. The secondary containment shall be as described in Section V-3.0 of the SAR.

C. Penetrations to the primary containment and piping passing through such