ATTACHMENT 6

Dresden Station Unit 2

DPR-19

Proposed Technical Specifications

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Proposed Technical Specifications

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NOTE: Page 81C-1 remains as is; it is not being deleted or revised.

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M. **Provisions to allow operation with one recirculation loop out of service:**

Am.

1. The steady-state thermal power level will not exceed 50% of rated

2. The Minimum Critical Power Ratio (MCPR) Safety Limit will be increased 0.03 (TS 1.1.A and 3.3.B.5.C)

3. The MCPR Limiting Condition for Operation (LCO) will be increased 0.03 (TS 3.5.K and Fig. 3.5-2)

4. The Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limits will be reduced to 70% of current values for all fuel types.

    (T.S. reference 3.5.I)

5. The APRM Scram and Rod Block Setpoints and the RBM Setpoints shall be reduced by 3.5% to read as follows:

    T.S. 2.1.A.1 $S \leq .58 \, WD + 58.5$
    T.S. 2.1.A.1* $S \leq (.58 \, WD + 58.5) \, FRP/MFLPD$
    T.S. 2.1.B $S \leq (.58 \, WD + 46.5)$
    T.S. 2.1.B* $S \leq (.58 \, WD + 46.5) \, FRP/MFLPD$
    T.S. 3.2.C (Table 3.2.3):
      APRM Upscale $\leq (.58 \, WD + 46.5) \, FR/MFLPD$
      RBM Upscale $\leq (.65 \, WD + 41.5)$

6. The suction valve in the idle loop is closed and electrically isolated until the idle loop is being prepared for return to service.

7. APRM flux noise will be measured once per shift and the recirculation pump speed will be reduced if the flux noise averaged over 1/2 hour exceeds 5% peak to peak, as measured on the APRM chart recorder.

8. The core plate delta p noise will be measured once per shift and the recirculation pump speed will be reduced if the noise exceeds 1 psi peak to peak.

* In the event that MFLPD exceeds FRP for General Electric fuel.

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

A.   (Deleted)

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.

C.   Critical Power Ratio (CPR) - The critical power ratio is the ratio of that assembly power which causes some point in the assembly to experience transition boiling to the assembly power at the reactor condition of interest as calculated by application of the XN-3 correlation. (Reference XN-NF-512)

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

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

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

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

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

I. Limiting Conditions for Operation (LCO) - The
limiting conditions for operation specify the
minimum acceptable levels of system perform-
ance necessary to assure safe startup and op-
eration of the facility. When these conditions
are met, the plant can be operated safely and
abnormal situations can be safely controlled.

J. Limiting Safety System Setting (LSSS) - The
limiting safety system settings are settings on
instrumentation which initiate the automatic
protective action at a level such that the safety
limits will not be exceeded. The region
between the safety limit and these settings
represents margin with normal operation lying
below these settings. The margin has been
established so that with proper operation of the
instrumentation the safety limits will never be
exceeded.

K. Fraction of Limiting Power Density (FLPD)
For fuel fabricated by GE, the fraction
of limiting power density is the ratio
of the Linear Heat Generation Rate (LHGR)
existing at a given location to the
design LHGR for that bundle type. FLPD
does not apply to ENC fuel.

L. Logic System Function Test - A logic sys-
tem functional test means a test of all relays
and contacts of a logic circuit from sensor
to activated device to insure all components
are operable per design intent. Where possi-
ble, action will go to completion, i.e., pumps
will be started and valves opened.

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

N. Mode - The reactor mode is that which is
established by the mode-selector-switch.

O. Operable - A system, subsystem, train, component, or device
shall be operable when it is capable of performing its
specified function(s). Implicit in this definition shall
be the assumption that all necessary attendant instrumen-
tation, controls, normal and emergency electrical power
sources, 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).

P. Operating - Operating means that a system, subsystem,
train, component or device is performing its intended
functions in its required manner.

Q. Operating Cycle - Interval between the end
of one refueling outage and the end of the
next subsequent refueling outage.

R. Primary Containment Integrity - Primary
containment integrity means that the drywell
and pressure suppression chamber are intact
and all of the following conditions are satisfied:

1. All manual containment isolation valves on
lines connecting to the reactor coolant sys-
tem or containment which are not required
to be open during accident conditions are
closed.

2. At least one door in each airlock is closed
and sealed.

3. All automatic containment isolation valves
are operable or deactivated in the isolated
position.

4. All blind flanges and manways are closed.

S. Protective Instrumentation Definitions

1. Instrument Channel - An instrument chan-
nel means an arrangement of a sensor and
auxiliary equipment required to generate
and transmit to a trip system a single trip
signal related to the plant parameter
monitored by that instrument channel.

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

J. Limiting Safety System Setting (LSSS) - The limiting safety system settings are settings on instrumentation which initiate the automatic protective action at a level such that the safety limits will not be exceeded. The region between the safety limit and these settings represents margin with normal operation lying below these settings. The margin has been established so that with proper operation of the instrumentation the safety limits will never be exceeded.

K. Fraction of Limiting Power Density (FLPD)
For fuel fabricated by GE, the fraction
of limiting power density is the ratio
of the Linear Heat Generation Rate (LHGR)
existing at a given location to the
design LHGR for that bundle type. FLPD
does not apply to ENC fuel.

L. Logic System Function Test - A logic system functional test means a test of all relays and contacts of a logic circuit from sensor to activated device to insure all components are operable per design intent. Where possible, action will go to completion, i.e., pumps will be started and valves opened.

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

N. Mode - The reactor mode is that which is established by the mode-selector-switch.

O. Operable - A system, subsystem, train, component, or device shall be operable when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electrical power sources, 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).

P. Operating - Operating means that a system, subsystem, train, component or device is performing its intended functions in its required manner.

Q. Operating Cycle - Interval between the end of one refueling outage and the end of the next subsequent refueling outage.

R. Primary Containment Integrity - Primary containment integrity means that the drywell and pressure suppression chamber are intact and all of the following conditions are satisfied:

1. All manual containment isolation valves on lines connecting to the reactor coolant system or containment which are not required to be open during accident conditions are closed.

2. At least one door in each airlock is closed and sealed.

3. All automatic containment isolation valves are operable or deactivated in the isolated position.

4. All blind flanges and manways are closed.

S. Protective Instrumentation Definitions

1. Instrument Channel - An instrument channel means an arrangement of a sensor and auxiliary equipment required to generate and transmit to a trip system a single trip signal related to the plant parameter monitored by that instrument channel.

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Amendment No.

1.1 SAFETY LIMIT	2.1 LIMITING SAFETY SYSTEM SETTING
1.1 FUEL CLADDING INTEGRITY **Applicability** The Safety Limits established to preserve the fuel cladding integrity apply to these 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. **Specifications** A. Reactor Pressure >800 psig and Core Flow > 10% of Rated. The existence of a minimum critical power ratio (MCPR) less than 1.05 shall constitute violation of the MCPR fuel cladding integrity safety limit.	**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. Neutron Flux Trip Settings The limiting safety system trip settings shall be as specified below:

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1. **APRM Flux Scram Trip Setting (Run Mode)**

   When the reactor mode switch is in the
   run position, the APRM flux scram setting
   shall be:

$$S \leq \left[.58W_D + 62\right]$$

   with a maximum set point of 120% for core
   flow equal to 98 x 10^6 lb/hr and greater,
   where:

   S - setting in per cent of rated power

   W_D - per cent of drive flow required to produce
          a rated core flow of 98 Mlb/hr.

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

   Where: $S \leq (.58W_D + 62)\left[\dfrac{FRP}{MFLPD}\right]$

   FRP = fraction of rated thermal power
         (2527) MWt)

   MFLPD = maximum fraction of limiting
           power density for GE fuel

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

1.1 SAFETY LIMIT	2.1 LIMITING SAFETY SYSTEM SETTING
	This adjustment may also be performed by increasing the APRM gain by the inverse ratio, MFLPD/FRP, which accomplishes the same degree of protection as reducing the trip setting by FRP/MFLPD. 2. APRM Flux Scram Trip Setting (Refuel or Startup and Hot Standby Mode) When the reactor mode switch is in the refuel startup/hot standby position, the APRM scram shall be set at less than or equal to 15% of rated neutron flux.

1.1 SAFETY LIMIT	2.1 LIMITING SAFETY SYSTEM SETTING

B. Core Thermal Power Limit (Reactor Pressure ≤ 800 psig)

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

C. Power Transient

1. The neutron flux shall not exceed the scram setting established in Specification 2.1.A for longer than 1.5 seconds as indicated by the process computer.

2. When the process computer is out of service, this safety limit shall be assumed to be exceeded if the neutron flux exceeds the scram setting established by Specification 2.1.A and a control rod scram does not occur.

D. Reactor Water Level (Shutdown Condition)

Whenever the reactor is in the shutdown condition with irradiated fuel in the reactor vessel, the water level shall not be less than that corresponding to 12 inches above the top of the active fuel*when it is seated in the core.

*Top of active fuel is defined to be 360 inches above vessel zero (see Bases 3.2).

3. IRM Flux Scram Trip Setting

The IRM flux scram setting shall be set at less than or equal to 120/125 of full scale.

B. APRM Rod Block Setting

The APRM rod block setting shall be:

$$S \leq \left[.58W_D + 50\right]$$

The definitions used above for the APRM scram trip apply.

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

$$S \leq (.58W_D + 50)\left[\frac{FRP}{MFLPD}\right]$$

The definitions used above for the APRM scram trip apply.

The ratio of FRP to MFLPD shall be set equal to 1.0 unless the actual operating value is less than 1.0. In which case the actual operating value will be used. The adjustment may also be performed by increasing the APRM gain by the inverse ratio, MFLPD/FRP, which accomplishes the same degree of protection as reducing the trip setting by FRP/MFLPD.

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FUEL CLADDING INTEGRITY

The fuel cladding integrity limit is set such that no calculated fuel damages would occur as a result of an abnormal operational transient. Because fuel damage is not directly observable, a step-back approach is used to establish a Safety Limit such that the minimum critical power ratio (MCPR) is no less than the MCPR fuel cladding integrity safety limit. MCPR > the MCPR fuel cladding integrity safety limit represents a conservative margin relative to the conditions required to maintain fuel cladding integrity by assuring that the fuel does not experience transition boiling.

The fuel cladding is one of the physical barriers which separate radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosions or use related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions and the protection system safety settings. While fission product migration from cladding perforation is just as measurable as that from use related cracking, the thermally caused cladding perforation signals a threshold, beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding

Safety Limit is defined with margin to the conditions which would produce onset of transition boiling, (MCPR of 1.0.) These conditions represent a significant departure from the condition intended by design for planned operation. The MCPR fuel cladding integrity Safety Limit assures that during normal operation and during anticipated operational occurrences, at least 99.9% of the fuel rods in the core do not experience transition boiling. See reference XN-NF-524.

A. Reactor Pressure > 800 psig and Core Flow > 10% of Rated

Onset of transition boiling results in a decrease in heat transfer from the clad and, therefore, elevated clad temperature and the possibility of clad failure. However, the existence of critical power, or boiling transition, is not a directly observable parameter in an operating reactor. Therefore, the margin to boiling transition is calculated from plant operating parameters such as core power, core flow, feedwater temperature, and core power distribution. The margin for each fuel assembly is characterized by the critical power ratio (CPR) which is the ratio of the bundle power which would produce onset of transition boiling divided by the actual bundle power. The minimum value of this ratio for any bundle in the core is the minimum critical power ratio (MCPR). It is assumed that the plant operation is controlled to the nominal protective setpoints via the instrumented variables. (Figure 2.1-3).

The MCPR Fuel Cladding Integrity Safety Limit assures sufficient conservatism in the operating MCPR limit that in the event of an anticipated operational occurrence from the limiting condition for operation, at

1.1.A Reactor Pressure > 800 psig and
      Core Flow > 10% of Rated. (cont'd)

least 99.9% of the fuel rods in the
core would be expected to avoid
boiling transition. The margin between
calculated boiling transition (MCPR=1.00)
and the MCPR Fuel Cladding Integrity
Safety Limit is based on a detailed
statistical procedure which considers
the uncertainties in monitoring the
core operating state. One specific
uncertainty included in the safety
limit is the uncertainty inherent
in the XN-3 critical power correlation.
Refer to XN-NF-524 for the methodology
used in determining the MCPR Fuel
Cladding Integrity Safety Limit.

The XN-3 critical power correlation
is based on a significant body of
practical test data, providing a
high degree of assurance that the
critical power as evaluated by the
correlation is within a small per-
centage of the actual critical power
being estimated. The assumed
reactor conditions used in defining
the safety limit introduce conserv-
atism into the limit because
boundingly high radial power peaking
factors and boundingly flat local
peaking distributions are used to
estimate the number of rods in
boiling transition. Still further
conservatism is induced by the
tendency of the XN-3 correlation
to overpredict the number of rods
in boiling transition. These
conservatisms and the inherent
accuracy of the XN-3 correlation
provide a reasonable degree of
assurance that during sustained
operation at the MCPR Fuel Cladding

Integrity Safety Limit there would be no trans-
ition boiling in the core. If boiling transition
were to occur, however, there is reason to
believe that the integrity of the fuel would
not necessarily be compromised. Significant
test data accumulated by the U. S. Nuclear
Regulatory Commission and private organizations
indicate that the use of a boiling transition
limitation to protect against cladding failure
is a very conservative approach; much of
the data indicates that LWR fuel can survive
for an extended period in an environment of
transition boiling.

If the reactor pressure should ever exceed
the limit of applicability of the XN-3
critical power correlation as defined in
XN-NF-512, it would be assumed that the
MCPR Fuel Cladding Integrity Safety Limit
had been violated. This applicability pressure
limit is higher than the pressure safety limit
specified in Specification 1.2. For fuel
fabricated by General Electric Company,
operation is further constrained to a maximum
linear heat generation rate (LHGR) of 13.4 kW/ft
by Specification 3.5.J. This constraint is
established to provide adequate safety margin
to 1% plastic strain for abnormal operational
transients initiated from high power conditions.
Specification 2.1.A.1 provides for equivalent
safety margin for transients initiated from
lower power conditions by adjusting the APRM
flow-biased scram by the ratio of FRP/MFLPD.
Specification 3.5.J establishes the maximum
value of LHGR which cannot be exceeded during
steady power operation for GE fuel types.

For fuel fabricated by Exxon Nuclear Company,
(ENC) fuel design criteria have been established
to provide protection against fuel centerline
melting and cladding strain, ENC has performed

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Safety Limit Bases

1.1.A . Reactor Pressure > 800 psig and
Core Fuel > 10% of Rated.   (cont'd)

fuel design analysis which demonstrate
that centerline melting is not predicted
to occur during transient overpower
conditions throughout the life of the
fuel, Protection of the MCPR and MAPLHGR
limits and operation within the power
distribution assumptions of the fuel
design analysis will provide adequate
protection against centerline melt and
ensures compliance with ENC's clad
overstrain criteria for steady state and
transient operation.   Since ENC's
design criteria are more conservative
than the 1% plastic strain limitation on
GE fuel, the LHGR limitation and APRM
scram adjustment for GE fuel established
in specifications 3.5.J and 2.1.A.1
respectively are unnecessary for the
protection of ENC fuel.   The procedural
controls of specification 3.1.B will
ensure that operation of ENC fuel remains
within the power distribution assumptions
of the fuel design analysis.

| 1.1 | Safety Limit Bases | 2.1 | Limiting Safety System Setting Bases |

1.1.C Power Transient (cont'd)

The computer provided has a
sequence annunciation program which
will indicate the sequence in which
scrams occur such as neutron flux,
pressure, etc. 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 normally will be
available for analyzing scrams; how-
ever, if the computer information should
not be available for any scram analysis,
Specification 1.1.C.2 will be relied on
to determine if a safety limit has been
violated.

During periods when the reactor is shut
down, consideration must also be given
to water level requirements due to the
effect of decay heat. If reactor water
level should drop below the top of the
active fuel during this time, the
ability to cool the core is reduced.
This reduction in core cooling cap-
ability could lead to elevated cladding
temperatures and clad perforation. The
core will be cooled sufficiently to pre-
vent clad melting should the water level
be reduced to two-thirds the core height.
Establishment of the safety limit at 12
inches above the top of the fuel* provides
adequate margin. This level will be con-
tinuously monitored whenever the recir-
culation pumps are not operating.

*Top of active fuel is defined to be
360 inches above vessel zero (see
Bases 3.2).

FUEL CLADDING INTEGRITY

The abnormal operational transients
applicable to operation of the units
have been analyzed throughout the
spectrum of planned operating con-
ditions up to the rated thermal power
condition of 2527 MWt. In addition,
2527 MWt is the licensed maximum steady-
state power level of the units. This
maximum steady-state power level will
never knowingly be exceeded. See reference
XN-NF-79-71.

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2.1   FUEL CLADDING INTEGRITY   (cont'd)

Conservatism is incorporated into the
transient analyses which define the
MCPR operating limits. Variables which
inherently possess little or no
uncertainty or whose uncertainty has
little or no effect on the outcome
of the limiting transient are selected
at bounding values. Variables which
possess significant uncertainty that
may have undesirable effects on thermal
margins are addressed statistically.
Statistical methods used in the
transient analyses are described in
XN-NF-81-22. The MCPR operating
limits are established such that the
occurrence of the limiting transient
will not result in the violation
of the MCPR Fuel Cladding Integrity
Safety Limit in at least 95% of the
random statistical combinations
of uncertainties. In general, the
variables with the greatest statistical
significance to the consequences of
anticipated operational occurrences are
the reactivity feedback associated with
the formation and removal of coolant
voids and the timing of the control
rod scram.

Steady-state operation without forced
reoirculation will not be permitted,
except during startup testing. The
analysis to support operation at
various power and flow relationships
has considered operation with either
one or two recirculation pumps.

The bases for individual trip settings
are discussed in the following para-
graphs.

For analyses of the thermal consequences of
the transients, the MCPR's stated in paragraph
3.5.K as the limiting condition of operation
bound those which are conservatively assumed
to exist prior to initiation of the transients.

A.   Neutron Flux Trip Settings

1.   APRM Flux Scram Trip Setting (Run Mode)

The average power range monitoring
(APRM) system, which is calibrated
using heat balance data taken during
steady-state conditions, reads in
percent of rated thermal power. Be-
cause 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. There-
fore, during abnormal operational
transients, the thermal power of the
fuel will be less than that indicated
by the neutron flux at the scram setting.
Analyses demonstrate that with a 120
percent scram trip setting, 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.

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Amendment No.

2.1.A. Neutron Flux Trip Settings

1. APRM Flux Scram Trip Setting
(Run Mode)   cont'd)

An increase in the APRM scram trip
setting would decrease the margin
present before the fuel cladding
integrity Safety Limit is reached.
The APRM scram trip setting was
determined by an analysis of margins
required to provide a reasonable
range for maneuvering during operation.
Reducing this operating margin would
increase the frequency of spurious
scrams which have an adverse effect
on reactor safety because of the
resulting thermal stresses. Thus,
the APRM scram trip setting was
selected because it provides adequate
margin for the fuel cladding integrity
Safety Limit yet allows operating
margin that reduces the possibility
of unnecessary scrams.

The scram trip setting must be
adjusted to ensure that the LHGR
transient peak for G.E. fuel
is not increased for any combin-
ation of Maximum Fraction of
Limiting Power Density (MFLPD)
and reactor core thermal power,
The scram setting is adjusted in
accordance with the formula in
specification 2.1.A.1. when the
MFLPD is greater than the fraction
of rated power (FRP).

The adjustment may also be
accomplished by increasing
the APRM gain by the reciprocal
of FRP/MFLPD. This provides the
same degree of protection as

reducing the trip setting by FRP/
MFLPD by raising the initial
APRM reading closer to the trip
setting such that a scram would be
received at the same point in a
transient as if the trip setting
had been reduced.

2. APRM Flux Scram Trip Setting
(Refuel or Start & Hot Standby Mode)

For operation in the startup mode while
the reactor is at low pressure, the APRM
scram setting of 15 percent of rated power
provides adequate thermal margin between the
the setpoint and the safety limit, 25 per-
cent of rated. The margin is adequate to
accommodate anticipated maneuvers associated
with power plant startup. Effects of in-
creasing pressure at zero or low void con-
tent are minor, cold water from sources
available during startup is not much colder
than that already in the system, tempera-
ture coefficients are small, and con-
trol rod patterns are constrained to
be uniform by operating procedures
backed up by the rod worth minimizer.
Of all possible sources of reactivity
input, uniform control rod withdrawal
is the most probable cause of signifi-
cant power rise. Because the flux
distribution associated with uniform
rod withdrawals does not involve high
local peaks, and because several rods
must be moved to change power by a
significant percentage of rated power,
the rate of power rise is very slow.
Generally, the heat flux is in near
equilibrium with the fission rate. In
an assumed uniform rod withdrawal ap-
proach to the scram level, the rate of
power rise is no more than 5 percent
of rated power per minute, and the
APRM system would be more than adequate 15

2.1.A. Neutron Flux Trip Setting

    2. APRM Flux Scram Trip Setting (Refuel
        or Start and Hot Standby Mode) (cont'd)

to assure a scram before the power
could exceed the safety limit. The 15
percent APRM scram remains active un-
til the mode switch is placed in the
RUN position. This switch occurs when
reactor pressure is greater than 850
psig.

3. **IRM Flux Scram Trip Setting**

The IRM system consists of 8 chambers,
4 in each of the reactor protection
system logic channels. The IRM is a
5-decade instrument which covers the
range of power level between that
covered by the SRM and the APRM. The
5 decades are broken down into 10 ranges,
each being one-half of a decade in size.

2.1.A. Neutron Flux Trip Setting

3. IRM Flux Scram Trip Setting (cont'd)

The IRM scram trip setting of 120 divisions is active in each range of the IRM. For example, if the instrument were on range 1, the scram setting would be a 120 divisions for that range; likewise, if the instrument were on range 5, the scram would be 120 divisions on that range. Thus, as the IRM is ranged up to accomodate the increase in power level, the scram trip setting is also ranged up.

The most significant sources of reactivity change during the power increase are due to control rod withdrawal. In order to ensure that the IRM provided adequate protection against the single rod withdrawal error, a range of rod withdrawal accidents was analyzed. This analysis included starting the accident at various power levels. The most severe case involves an initial condition in which the reactor is just subcritical and the IRM system is not yet on scale.

Additional conservatism was taken in this analysis by assuming that the IRM channel closest to the withdrawn rod is bypassed. The results of this analysis show that the reactor is scrammed and peak power limited to one percent of rated power, thus maintaining MCPR above the MCPR fuel cladding integrity safety limit. Based on the above analysis, the IRM provides protection against local control rod withdrawal errors and continuous withdrawal of control rods in sequence and provides backup protection for the APRM.

2.1.B APRM Rod Block Trip Setting

Reactor power level may be varied by moving control rods or by varying the recirculation flow rate. The APRM system provides a control rod block to prevent gross rod withdrawal at constant recirculation flow rate to protect against grossly exceeding the MCPR fuel cladding integrity safety limit. This rod block trip setting, which is automatically varied with recirculation loop flow rate, prevents an increase in the reactor power level to excessive values due to control rod withdrawal. The flow variable trip setting provides substantial margin from fuel damage, assuming a steady-state operation at the trip setting, over the entire recirculation flow range. The margin to the Safety Limit increases as the flow decreases for the specified trip setting versus flow relationship; therefore the worst case MCPR which could occur during steady-state operation is at 100% of rated thermal power because of the APRM rod block trip setting. The actual power distribution in the core is established by specified control rod sequences and is monitored continuously by the in-core LPRM system. As with the APRM scram trip setting, the APRM rod block trip setting is adjusted downward or APRM gain increased if the maximum fraction of limiting power density for G.E. fuel exceeds the fraction of rated power, thus preserving the APRM rod block safety margin.

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E. Turbine Stop Valve Scram - The turbine stop valve closure scram trip anticipates the pressure, neutron flux and heat flux increase that could result from rapid closure of the turbine stop valves. With a scram trip setting of 10 percent of valve closure from full open, the resultant increase in surface heat flux is limited such that MCPR remains above the MCPR fuel cladding integrity safety limit, even during the worst case transient that assumes the turbine bypass is closed.

F. Generator Load Rejection Scram - The generator load rejection scram is provided to anticipate the rapid increase in pressure and neutron flux resulting from fast closure of the turbine control valves due to a load rejection and subsequent failure of the bypass; i.e., it prevents MCPR from becoming less than the MCPR fuel cladding integrity safety limit for this transient. For the load rejection without bypass transient from 100% power, the peak heat flux (and therefore LHGR) increases on the order of 15% which provides wide margin to the value corresponding to fuel centerline melting and 1% cladding strain.

G. Reactor Coolant Low Pressure Initiates Main Steam Isolation Valve Closure - The low pressure isolation at 850 psig was provided to give protection against fast reactor depressurization and the resulting rapid cooldown of the vessel. Advantage was taken of the scram feature which occurs when the main steam line isolation valves are closed to provide for reactor shutdown so that operation at pressures lower than those specified in the thermal hydraulic safety limit does not occur, although operation at a pressure lower than 850 psig would not necessarily constitute an unsafe condition.

H. Main Steam Line Isolation Valve Closure Scram - The low pressure isolation of the main steam lines at 850 psig was provided to give protection against rapid reactor depressurization and the resulting rapid cooldown of the vessel. Advantage was taken of the scram feature which occurs when the main steam line isolation valves are closed, to provide for reactor shutdown so that high power operation at low reactor pressure does not occur, thus providing protection for the fuel cladding integrity safety limit. Operation of the reactor at pressures lower than 850 psig requires that the reactor mode switch be in the startup position where protection of the fuel cladding integrity safety limit is provided by the IRM high neutron flux scram. Thus, the combination of main steam line low pressure isolation and isolation valve closure scram assures the availability of neutron flux scram protection over the entire range of applicability of the fuel cladding integrity safety limit. In addition, the isolation valve closure scram anticipates the pressure and flux transients which occur during normal or inadvertent isolation valve closure. With the scrams set at 10% valve closure, there is no appreciable increase in neutron flux.

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1.2 SAFETY LIMIT	2.2 LIMITING SAFETY SYSTEM SETTING
1.2 REACTOR COOLANT SYSTEM	2.2 REACTOR COOLANT SYSTEM
Applicability:	Applicability:
Applies to limits on reactor coolant system pressure.	Applies to trip settings of the instruments and devices which are provided to prevent the reactor system safety limits from being exceeded.
Objective:	Objective:
To establish a limit below which the integrity of the reactor coolant system is not threatened due to an overpressure condition.	To define the level of the process variables at which automatic protective action is initiated to prevent the safety limits from being exceeded.
Specification:	Specification:
The reactor coolant system pressure shall not exceed 1345 psig at any time when irradiated fuel is present in the reactor vessel.	A. Reactor Coolant High Pressure Scram shall be ≤1060 psig.
	B. Primary System Safety Valve Nominal Settings shall be as follows:
	1 valve at 1115 psig* 2 valves at 1240 psig 2 valves at 1250 psig 2 valves at 1260 psig 2 valves at 1260 psig
	The allowable setpoint error for each valve shall be +1%.
	*Target Rock combination safety/relief valve

1.2 **The reactor coolant system integrity is an impor-
tant barrier in the prevention of uncontrolled re-
lease of fission products. It is essential that the
integrity of this system be protected by establishing
a pressure limit to be observed for all operating
conditions and whenever there is irradiated fuel in
the reactor vessel.**

The pressure safety limit of 1345 psig as measured
by the vessel steam space pressure indicator ensures
margin to 1375 psig at the lowest elevation of the
reactor vessel. The 1375 psig value is derived from
the design pressures of the reactor pressel vessel
and coolant system piping. The respective design
pressures are 1250 psig at 575°F and 1175 psig at
560°F. The pressure safety limit was chosen as
the lower of the pressure transients permitted by
the applicable design codes: ASME Boiler and
Pressure Vessel Code, Section III for the pressure
vessel and USASI Code for the reactor coolant system
piping. The ASME Boiler and Pressel Vessel Code
permits pressure transients up to 10% power over
design pressure (110% x 1250 - 1375 psig), and the
USASI Code permits pressure transients up to 20%
over the design pressure (120% x 1175 = 1410 psig).
The Safety limit pressure of 1375 psig is referenced
to the lowest elevation of the reactor vessel.

The design pressure for the recirc. suction
line piping (1175psig) was chosen relative
to the reactor vessel design pressure.
Demonstrating compliance of the peak
vessel pressure with the ASME overpressure
protection limit (1375psig) assures
compliance of the suction piping with
the USASI limit (1410psig). Evaluation
methodology used to assure that this
safety limit pressure is not exceeded
for any reload is documented in Reference
XN-NF-79-71. The design basis for the
reactor pressure vessel makes evident the
substantial margin of protection against
failure at the safety pressure limit of
1375 psig. The vessel has been designed
for a general membrane stress no greater

than 26,700 psi at an internal pressure
of 1250 psig: this is a factor of 1.5
below the yield strength of 40,100 psi
at 575°F. At that pressure limit of
1375 psig, the general membrane stress
will only be 29,400 psi, still safely
below the yield strength.

The relationships of stress levels to
yield strength are comparable for the
primary system piping and provide a
similar margin of protection at the
established safety pressure limit.

The normal operating pressure of the
reactor coolant system is 1000psig.
For the turbine trip or loss of
electrical load transients, the turbine
trip scram or generator load rejection
scram, together with the turbine bypass
system, limit the pressure to approximately
1100psig (2). In addition, pressure
relief valves have been provided to
reduce the probability of the safety
valves, which discharged to the drywell,
operating in the event that the turbine
bypass should fail.

Finally, the safety valves are sized
to keep the reactor vessel peak pressure
below 1375 psig with no credit taken
for the relief valves during the
postulated full closure of all MSIV's
without direct (valve position switch)
scram. Credit is taken for the neutron
flux scram, however,

The indirect flux scram and safety valve
actuation provide adequate margin
below the peak allowable vessel pressure
of 1375 psig.

Reactor pressure is continuously monitored
in the control room during operation on
a 1500 psi full scale pressure recorder.

(4) SAR, Section 11.2.2 -
also: "Dresden 3 Second Reload License
Submittal," 9-14-73
also: "Dresden Station Special Report
No. 29 Supplement B."

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Bases:

2.2 In compliance with Section III of the ASME Code, the
safety valves must be set to open at no higher than
103% of design pressure, and they must limit the
reactor pressure to no more than 110% of design
pressure. Both the neutron flux scram and safety
valve actuation are required to prevent overpres-
surizing the reactor pressure vessel and thus
exceeding the pressure safety limit. The pressure
scram is available as a backup protection
to the direct valve position trip scrams
and the high flux scram.

If the high flux scram
were to fail, a high pressure scram would
occur at 1060 psig. Analyses are performed
as described in reference XN-NF-79-71 for
each reload to assure that the pressure safety
limit is not exceeded.

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3.1 LIMITING CONDITION FOR OPERATION	4.1 SURVEILLANCE REQUIREMENT

3.1 REACTOR PROTECTION SYSTEM

Applicability:

Applies to the instrumentation and associated devices which initiate a reactor scram.

Objective:

To assure the operability of the reactor protection system.

Specification:

A. The setpoints, minimum number of trip systems, and minimum number of instrument channels that must be operable for each position of the reactor mode switch shall be as given in Table 3.1.1. The system response times from the opening of the sensor contact up to and including the opening of the trip actuator contacts shall not exceed 50 milliseconds.

B. If during operation, the maximum fraction of limiting power density for fuel fabricated by GE exceeds the fraction of rated power when operating above 25% rated thermal power, either:

- a. The APRM scram and rod block settings shall be reduced to the values given by the equations in Specifications 2.1.A.1 and 2.1.B.

4.1 REACTOR PROTECTION SYSTEM

Applicability:

Applies to the surveillance of the instrumentation and associated devices which initiate reactor scram.

Objective:

To specify the type and frequency of surveillance to be applied to the protection instrumentation.

Specification:

A. Instrumentation systems shall be functionally tested and calibrated as indicated in Tables 4.1.1 and 4.1.2, respectively.

B. Daily during reactor power operation above 25% rated thermal power, the core power distribution shall be checked for:

1. Maximum fraction of limiting power density for fuel fabricated by GE (MFLPD) and compared with the fraction of rated power (FRP).

2. For compliance with assumptions of the Fuel Design Analysis of overpower conditions for fuel fabricated by ENC.

3.1 LIMITING CONDITIONS FOR OPERATION   4.1 SURVEILLANCE REQUIREMENT

Specifications (cont'd)

b. The power distribution shall be changed
    such that the maximum fraction of limiting
    power density no longer exceeds the fraction
    of rated power.

    For fuel fabricated by ENC, operation of
    the core shall be limited to ensure the
    power distribution is consistent with that
    assumed in the Fuel Design Analysis for over-
    power conditions.

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a half scram and rod block condition. Thus, if the calibration were performed during operation, flux shaping would not be possible. Based on experience at other generating stations, drift of instruments, such as those in the Flow Biasing Nework, is not significant and therefore, to avoid spurious scrams, a calibration frequency of each refueling outage is established.

Group (C) devices are active only during a given portion of the operational cycle. For example, the IRM is active during startup and inactive during full-power operation. Thus, the only test that is meaningful is the one performed just prior to shutdown or startup; i.e., the tests that are performed just prior to use of the instrument.

Calibration frequency of the instrument channel is divided into two groups. These are as follows:

1. Passive type indicating devices that can be compared with like units on a continuous basis.

2. Vacuum tube or semiconductor devices and detectors that drift or lose sensitivity.

Experience with passive type instruments in Commonwealth Edison generating stations and substations indicates that the specified calibrations are adequate. For those devices which employ amplifiers, etc., drift specifications call for drift to be less than 0.4%/month; i.e., in the period of a month a drift of .4% would occur and thus providing for adequate margin.

For the APRM system drift of electronic apparatus is not the only consideration in determining a calibration frequency. Change in power distribution and loss of chamber sensitivity dictate a calibration every seven days. Calibration on this frequency assures plant operation at or below thermal limits.

A comparison of Tables 4.1.1 and 4.1.2 indicates that six instrument channels have not been included in the latter Table. These are: Mode Switch in Shutdown, Manual Scram, High Water Level in Scram Discharge Tank, Main Steam Line Isolation Valve Closure, Generator Load Rejection, and Turbine Stop Valve Closure. All of the devices or sensors associated with these scram functions are simple on-off switches and, hence, calibration is not applicable, i.e., the switch is either on or off. Further, these switches are mounted solidly to the device and have a very low probability of moving, e.g. the switches in the scram discharge volume tank. Based on the above, no calibration is required for these six instrument channels.

B. The MFLPD for fuel fabricated by GE shall be checked once per day to determine if the APRM gains or scram requires adjustment. This may normally be done by checking the LPRM readings, TIP traces, or process computer calculations. Only a small number of control rods are moved daily and thus the peaking factors are not expected to change significantly and thus a daily check of the MFLPD is adequate.

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

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3.2 LIMITING CONDITION FOR OPERATION	4.2 SURVEILLANCE REQUIREMENT

C. Control Rod Block Actuation

1. The limiting conditions of operation for the instrumentation that initiates control rod block are given in Table 3.2.3.

2. The minimum number of operable instrument channels specified in Table 3.2.3 for the Rod Block Monitor may be reduced by one in one of the trip systems for maintenance and/or testing, provided that this condition does not last longer than 24 hours in any 30-day period. In addition, one channel may be bypassed above 30% rated power without a time restriction provided that a limiting control rod pattern does not exist and the remaining RBM channel is operable.

D. Steam Jet-Air Ejector Off Gas System

1. Except as specified in 3.2.D.2. below, both steam-jet air ejector off-gas system radiation monitors shall be operable during reactor power operation. The trip settings for the monitors shall be set at a value not to exceed the equivalent of the stack release limit specified in Specification 3.8. The time delay setting for closure of the steam jet-air ejector isolation valves shall not exceed 15 minutes.

3.2 LIMITING CONDITION FOR OPERATION	4.2 SURVEILLANCE REQUIREMENT

2. From and after the date that one of the two steam-jet air ejector off-gas system radiation monitors is made or found to be inoperable, continued reactor power operation is permissible during the next seven days provided the inoperable monitor is tripped in the upscale position.

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INSTRUMENTATION THAT INITIATES ROD BLOCK

Table 3.2.3

Minimum No. of Operable Inst. Channels Per Trip System(1)	Instrument	Trip Level Setting
1	APRM upscale (flow bias)(7)	$\leq \left[0.58W_D + 50 \dfrac{FRP}{MFLPD} \right]$ (2)
1	APRM upscale (refuel and Startup/Hot Standby mode)	$\leq 12/125$ full scale
2	APRM downscale (7)	$\geq 3/125$ full scale
1	Rod block monitor upscale (flow bias)(7)	$\leq 0.65\ W_D + 45$ (2)
1	Rod block monitor downscale (7)	$\geq 5/125$ full scale
3	IRM downscale (3)	$\geq 5/125$ full scale
3	IRM upscale	$\leq 108/125$ full scale
3	IRM detector not fully inserted in the core	
2(5)	SRM detector not in startup position	(4)
2(5)(6)	SRM upscale	$\leq 10^5$ counts/sec
1	Scram Discharge Volume Water Level - High	25 gal.

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TABLE 2.3 (Cont.)

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NOTES:

1. For the Startup/Hot Standby and Run positions of the Reactor Mode Selector Switch, there shall be two operable or tripped trip systems for each function, except the SRM rod blocks, IRM upscale, IRM downscale and IRM detector not fully inserted in the core need not be operable in the "Run" position and APRM downscale, APRM upscale (flow bias), and RBM downscale need not be operable in the Startup/Hot Standby mode. The RBM upscale need not be operable at less than 30% rated thermal power. One channel may be bypassed above 30% rated thermal. power provided that a limiting control rod pattern does not exist. For systems with more than one channel per trip system, if the first column cannot be met for both trip systems, the systems shall be tripped. For the Scram Discharge Volume water level high rod block, there is one instrument channel for one trip system.

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

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

4. This function may be bypassed when the count rate is ≥ 100 cps.

5. One of the four SRM inputs may be bypassed.

6. This SRM function may be bypassed in the higher IRM ranges when the IRM upscale rod block is operable.

7. Not required while performing low power physics tests at atmospheric pressure during or after refueling at power levels not to exceed 5 MW(t).

Bases:

3.2 In addition to reactor protection instrumentation which initiates a reactor scram, protective instrumentation has been provided which initiates action to mitigate the consequences of accidents which are beyond the operators ability to control, or terminates operator errors before they result in serious consequences. This set of Specifications provides the limiting conditions of operation for the primary system isolation function, initiation of the emergency core cooling system, control rod block and standby gas treatment systems. The objectives of the specifications are (i) to assure the effectiveness of the protective instrumentation when required by preserving its capability to tolerate a single failure of any component of such systems even during periods when portions of such systems are out of service for maintenance, and (ii) to prescribe the trip settings required to assure adequate performance. When necessary, one channel may be made inoperable for brief intervals to conduct required functional tests and calibrations.

Some of the settings on the instrumentation that initiates or controls core and containment cooling have tolerances explicitly stated where the high and low values are both critical and may have a substantial effect on safety. It should be noted that the setpoints of other instrumentation, where only the high or low end of the setting has a direct bearing on safety, are chosen at a level away from the normal operating range to prevent inadvertent actuation of the safety system involved and exposure to abnormal situations.

Isolation valves are installed in those lines that penetrate the primary containment and must be isolated during a loss of coolant accident so that the radiation dose limits are not exceeded during an accident condition. Actuation of these valves is initiated by protective instrumentation shown in

Table 3.2.1 which senses the conditions for which isolation is required. Such instrumentation must be available whenever primary containment integrity is required. The objective is to isolate the primary containment so that the guidelines of 10 CFR 100 are not exceeded during an accident.

The instrumentation which initiates primary system isolation is connected in a dual bus arrangement. Thus, the discussion given in the bases for Specification 3.1 is applicable here.

The low-reactor water level instrumentation is set to trip at >0 inches on the level instrument (top of active fuel is defined to be 360 inches above vessel zero) and after allowing for the full power pressure drop across the steam dryer the low level trip is at 504 inches above vessel zero, or 144 inches above the top of active fuel. Retrofit 8x8 fuel has an active fuel length 1.24 inches longer than earlier fuel designs. However, present trip setpoints were used in the LOCA analyses. This trip initiates closure of Group 2 and 3 primary containment isolation valves but does not trip the recirculation pumps (reference SAR Section 7.7.2). For a trip setting of 504 inches above vessel zero (144 inches above top of active fuel) and a 60-second valve closure time, the valves will be closed before perforation of the cladding occurs even for the maximum break; the setting is therefore adequate.

The low low reactor level instrumentation is set to trip when reactor water level is 444 inches above vessel zero (with top of active fuel defined as 360 inches above vessel zero, -39 inches is 84 inches above the top of active fuel).

This trip initiates closure of Group 1 primary containment isolation valves, Ref. Section 7.7.2.2 SAR, and also activates the ECC subsystems, starts the emergency diesel generator and trips the recirculation pumps. This trip setting level was chosen to be high enough to prevent spurious operation but low enough to initiate ECCS operation and primary system isolation so that no melting of the fuel cladding will occur and so that post accident cooling can be accomplished and the guidelines of 10 CFR 100 will not be violated. For the complete circumferential break of a 28-inch recirculation line and with the trip setting given above, ECCS initiation and primary system isolation are initiated in time to meet the above criteria.

3.3 LIMITING CONDITION FOR OPERATION	4.3 SURVEILLANCE REQUIREMENTS

C. Scram Insertion Times

   1. The average scram insertion time, based on the de-energization of the scram pilot valve solenoids as time zero, of all operable control rods in the reactor power operation condition shall be no greater than:

% Inserted From Fully Withdrawn	Avg. Scram Insertion Times (sec)
5	0.375
20	0.900
50	2.00
90	3.50

   The average of the scram insertion times for the three fastest control rods of all groups of four control rods in a two by two array shall be no greater than:

% Inserted From Fully Withdrawn	Avg. Scram Insertion Times (sec)
5	0.398
20	0.954
50	2.120
90	3.800

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

C. Scram Insertion Times

   1. After each refueling outage and prior to power operation with reactor pressure above 800 psig, all control rods shall be subject to scram-time tests from the fully withdrawn position. The scram times shall be measured without reliance on the control rod drive pumps.

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

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

indicative of a generic control rod drive
problem and the reactor will be shutdown.
Also, if damage within the control rod
drive mechanism and, in particular, cracks
in drive internal housings, cannot be ruled
out, then a generic problem affecting a
number of drives cannot be ruled out. Cir-
cumferential cracks resulting from stress
assisted intergranular corrosion have
occurred in the collet housing of drives at
several BWR's. This type of cracking could
occur in a number of drives and if the
cracks propagated until severance of the
collet housing occurred, scram could be pre-
vented in the affected rods. Limiting the
period of operation with a potentially
severed collet housing and requiring in-
creased surveillance after detecting one stuck
rod will assure that the reactor will not be
operated with a large number of rods with
failed collet housings.

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

B. Control Rod Withdrawal

1. Control rod dropout accidents as discussed in Ref-
erence XN-NF-80-19, Vol. 1, can lead to significant core
damage. If coupling integrity is maintained,
the possibility of a rod dropout accident is
eliminated. The overtravel position feature

provides a positive check as only uncoupled
drives may reach this position. Neutron in-
strumentation response to rod movement pro-
vides a verification that the rod is following
its drive. Absence of such response to drive
movement would provide cause for suspecting a
rod to be uncoupled and stuck. Restricting re-
coupling verifications to power levels above
20% provides assurance that a rod drop during
a recoupling verification would not result in a
rod drop accident.

2. The control rod housing support restricts the
outward movement of a control rod to less than
3 inches in the extremely remote event of a
housing failure. The amount of reactivity which
could be added by this small amount of rod with-
drawal, which is less than a normal single with-
drawal increment, will not contribute to any
damage to the primary coolant system. The design
basis is given in Section 6.6.1 of the SAR, and
the design evaluation is given in Section 6.6.3.
This support is not required if the reactor
coolant system is at atmospheric pressure since
there would then be no driving force to rapidly
eject a drive housing. Additionally, the support
is not required if all control rods are fully in-
serted, and if an adequate shutdown margin with one
control rod withdrawn has been demonstrated since
the reactor would remain subcritical even in the
event of complete ejection of the strongest con-
trol rod.

3. Control rod withdrawal and insertion sequences
are established to assure that the maximum in-
sequence individual control rod or control rod
sequences which are withdrawn could not be worth
enough to cause the rod drop accident design limit of
280 cal/gm to be exceeded if they were to drop out of
the core in the manner defined for the Rod Drop Accident.
These sequences are developed prior to initial
operation

of the unit following any refueling outage
and the requirement that an operator follow
these sequences is backed up by the opera-
ation of the RWM or a second qualified
station employe. These sequences are
developed to limit reactivity worths of
control rods and, together with the integral
rod velocity limiters and the action of the
control rod drive system, limit potential re-
activity insertion such that the results of
a control rod drop accident will not exceed
a maximum fuel energy content of 280 cal/gm.
The peak fuel enthalpy of 280 cal/gm is below
the energy content, 425 cal/gm, at which rapid
fuel dispersal and primary system damage have
been found to occur based on experimental data
as is discussed in Reference 1.

The analysis of the control rod drop accident
was originally presented in Sections 7.9.3,
14.2.1.2 and 14.2.1.4 of the Safety Analysis
Report. Improvements in analytical capability
have allowed a more refined analysis of the
control rod drop accident.

Bases(cont'd)

Parametric Control Rod Drop Accident analyses
have shown that for wide ranges of key reactor
parameters (which envelope the operating ranges
of these variables), the fuel enthalpy rise
during a postulated control rod drop accident
remains considerably lower than the 280 cal/gm
limit. For each operating cycle, cycle-specific
parameters such as maximum control rod worth,
Doppler coefficient effective delayed neutron
fraction and maximum four-bundle local peaking
factor are compared with the results of the
parametric analyses to determine the peak fuel
rod enthalpy rise. This value is then compared
against the Technical Specification limit
of 280 cal/gm to demonstrate compliance for
each operating cycle. If cycle specific values
of the above parameters are outside the range
assumed in the parametric analyses, an extension
of the analysis or a cycle specific analysis
may be required. Conservatism present in the
analysis, results of the parametric studies,
and a detailed description of the methodology
for performing the Control Rod Drop Accident
analysis are provided in reference XN-NF-80-19,
Volume 1 (Supplements 1 and 2).

Bases (con'd)

The Rod Worth Minimizer provides automatic
supervision to assure that out of sequence
control rods will not be withdrawn or inserted;
i.e., it limits operator deviations from planned
withdrawal sequences. Ref. Section 7.9 SAR.
It serves as a backup to procedural control of
control rod worth. In the event that the Rod
Worth Minimizer is out of service, when required,
a licensed operator or other qualified
technical employee can manually fulfill the
control rod pattern conformance functions of the
Rod Worth Minimizer. In this case, procedural
control is exercised by verifying all control
rod positions after the withdrawal of each
group, prior to proceeding to the next
group. Allowing substitution of a second
independent operator or engineer in case
of RWM inoperability recognizes the capability
to adequately monitor proper rod sequencing in
an alternate manner without unduly restrict-
ing plant operations. Above 20% power, there is
no requirement that the RWM be operable since
the control rod drop accident with out-of-
sequence rods will result in a peak fuel
energy content of less than 280 cal/gm. To
assure high RWM availability, the RWM is
required to be operating during a startup
for the withdrawal of a significant number
control rods for any startup after June 1, 1974.

4. The Source Range Monitor (SRM) system performs
no automatic safety system function; i.e., it
has no scram function. It does provide the

operator with a visual indication of neutron level. This is needed for knowledgeable and efficient reactor startup at low neutron level. The consequences of reactivity accidents are functions of the initial neutron flux. The requirement of at least 3 counts per second assures that any transient, should it occur, begins at or above the initial value of 10^{-8} of rated power used in the analyses of transients from cold conditions. One operable SRM channel would be adequate to monitor the approach to criticality using homogeneous patterns of scattered control rod withdrawal. A minimum of two operable SRM's are provided as an added conservatism.

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 rods according to a written sequence. The specified restrictions with one channel out of service conservatively assure that fuel damage will not occur due to rod withdrawal errors when this condition exists. Amendments 17/18 and 19/20 present the results of an evaluation of a rod block monitor failure. These amendments show that during reactor operation with certain limiting control rod patterns, the withdrawal of a designated single control rod could result in one or more fuel rods with MCPRs less than the MCPR fuel cladding integrity safety limit. During use of such patterns, it is judged that testing of the RBM system prior to withdrawal of such rods to assure its operability will assure that improper withdrawal does not occur. It is the responsibility of the Nuclear Engineer to identify these limiting patterns and the designated rods either when the patterns are initially established or as they develop due to the occurrence of inoperable control rods in other than limiting patterns.

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Scram Insertion Times

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

In the analytical treatment of most transients, 290 milliseconds are allowed between a neutron sensor reaching the scram point and the start of motion of the control rods. This is adequate and conservative when compared to the typically observed time delay of about 210 millisecons. Approximately 90 milliseconds after neutron flux reaches the trip point, the pilot scram value solenoid de-energizes and 120 milliseconds later the control rod motion is estimated to actually begin. However, 200 milliseconds rather than 120 milliseconds is conservatively assumed for this time interval in the transient analyses, and is also included in the allowable scram insertion times specified in Specification 3.3.C. In the statistical treatment of the limiting transients, a statistical distribution of total scram delay is used rather than the bounding value described above.

The performance of the individual control rod drives is monitored to assure that scram performance is not degraded. Fifty percent of the control rod drives in the reactor are tested every sixteen weeks to verify adequate performance. Observed plant data were used to determine the average scram performance used in the transient analyses,

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Scram Insertion Times (cont'd)

and the results of each set of control rod
scram tests during the current cycle are
compared against earlier results to verify
that the performance of the control rod
insertion system has not changed signifi-
cantly. If an individual test or group
of tests should be determined to fall outside
of the statistical population defining
the scram performance characteristics
used in the transient analyses, a re-
determination of thermal margin require-
ments is undertaken (as required by
Specification 3.5.K) unless it can be
shown that the number of individual drives
falling outside the statistical population
defining the nominal performance is
less than the allowable number of inoper-
able control rod drives. If the number
of statistically aberrant drives falls
within this limitation, operation will
be allowed to continue without rede-
termination of thermal margin require-
ments provided the identified aberrant
drives are fully inserted into the core
and deenergized in the manner of an
inoperable rod drive.

The scram times for all control rods
are measured at the time of each refueling
outage. Experience with the plant has
shown that control drive insertion
times vary little through the operating
cycle; hence no reassessment of thermal
margin requirements is expected under
normal conditions. The history of
drive performance accumulated to date
indicates that the 90% insertion times of
new and overhauled drives approximate a normal
distribution about the mean which tends to become
skewed toward longer scram times as operating
time is accumulated. The probability of a drive
not exceeding the mean 90% insertion time by
0.75 second is greater than 0.999 for a
normal distribution.

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

D. Control Rod Accumulators

The basis for this specification was not des-
cribed in the SAR and, therefore, is presented
in its entirety. Requiring no more than one
inoperable accumulator in any nine-rod square
array is based on a series of XY PDQ-4 quarter
core calculations of a cold, clean core. The
worst case in a nine-rod withdrawal sequence
resulted in a $k_{eff} < 1.0$ -- other repeating rod
sequences with more rods withdrawn resulted
in $k_{eff} > 1.0$. At reactor pressures in excess
of 800 psig, even those control rods with in-
operable accumulators will be able to meet re-
quired scram insertion times due to the action
of reactor pressure. In addition, they may be
normally inserted using the control-rod-drive
hydraulic system. Procedural control will
assure that control rods with inoperable accu-
mulators will be spaced in a one-in-nine array
rather than grouped together.

E. Reactivity Anomalies

During each fuel cycle excess operating reac-

G. Economic Generation Control System

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

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

(Revised with Changes 27 and 18 issued 3/29/74)

3.5 LIMITING CONDITION FOR OPERATION	4.5 SURVEILLANCE REQUIREMENT

D. Automatic Pressure Relief Subsystems

1. Except as specified in 3.5.D.2 and 3 below, the Automatic Pressure Relief Subsystem shall be operable whenever the reactor pressure is greater than 90 psig and irradiated fuel is in the reactor vessel.

2. From and after the date that one of the five relief valves of the automatic pressure relief subsystem is made or found to be inoperable when the reactor is pressurized above 90 psig with irradiated fuel in the reactor vessel, reactor operation is permissible only during the succeeding seven days unless repairs are made and provided that during such time the HPCI Subsystem is operable.

3. From and after the date that more than one of five relief valves of the automatic pressure relief subsystem made or found to be inoperable when the reactor is pressurized above 90 psig with irradiated fuel in the reactor vessel, reactor operation is permissible only during the succeeding 24 hours unless repairs are made and provided that during such time the HPCI Subsystem is operable.

D. Surveillance of the Automatic Pressure Relief Subsystem shall be performed as follows:

1. During each operating cycle the following shall be performed:

    a. A simulated automatic initiation which opens all pilot valves, and

    b. With the reactor at pressure each relief valve shall be manually opened. Relief valve opening shall be verified by a compensating turbine bypass valve or control valve closure.

    c. A logic system functional test shall be performed each refueling outage.

2. When it is determined that one relief valve of the automatic pressure relief subsystem is inoperable, the HPCI shall be demonstrated to be operable immediately and weekly thereafter

3. When it is determined that more than one relief valve of the automatic pressure relief subsystem is inoperable, the HPCI subsystem shall be demonstrated to be operable immediately

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3.5 LIMITING CONDITION FOR OPERATION	4.5 SURVEILLANCE REQUIREMENT

I. Average Planar LHGR

During steady state power operation, the Average Planar Linear Heat Generation Rate (APLHGR) of all the rods in any fuel assembly, as a function of average planar exposure for G.E. fuel and average bundle exposure for Exxon fuel at

any axial location, shall not exceed the maximum average planar LHGR shown in Figure 3.5-1. If at any time during operation it is determined by normal surveillance that the limiting value for APLHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the APLHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

I. Average Planar Linear Heat Generation Rate (APLHGR)

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

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3.5 LIMITING CONDITION FOR OPERATION	4.5 SURVEILLANCE REQUIREMENT

3.5.J <u>LOCAL LHGR</u>

During steady state power operation, the linear heat generation rate (LHGR) of any rod in any fuel assembly fabricated by GE at any axial location shall not exceed the design value of 13.4 kw/ft.

If at any time during operation, it is determined by normal surveillance that the limiting value for LHGR for G.E. fuel is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the LHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits

J. <u>Linear Heat Generation Rate (LHGR)</u>

The LHGR shall be checked daily during reactor operation at $\geq$ 25% rated thermal power.

DIETZGEN CORPORATION
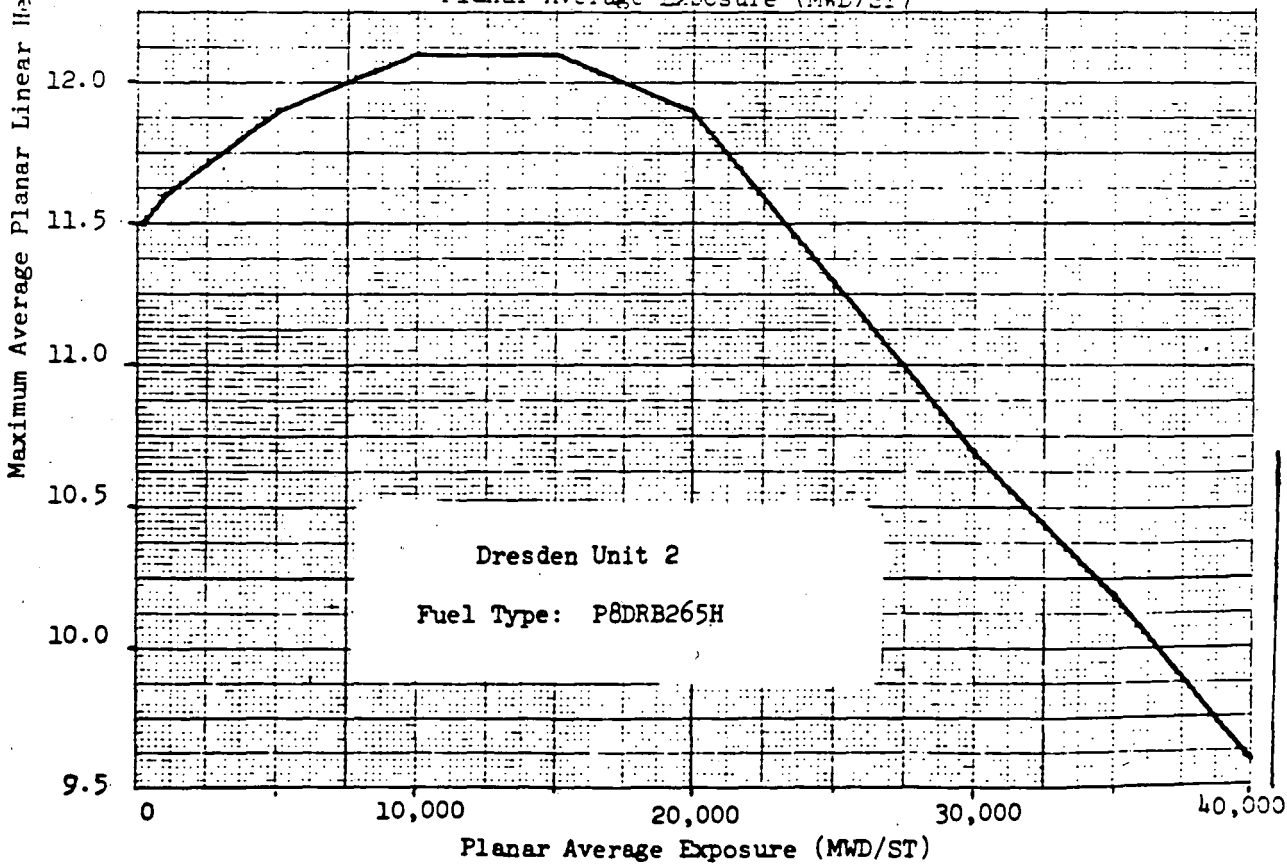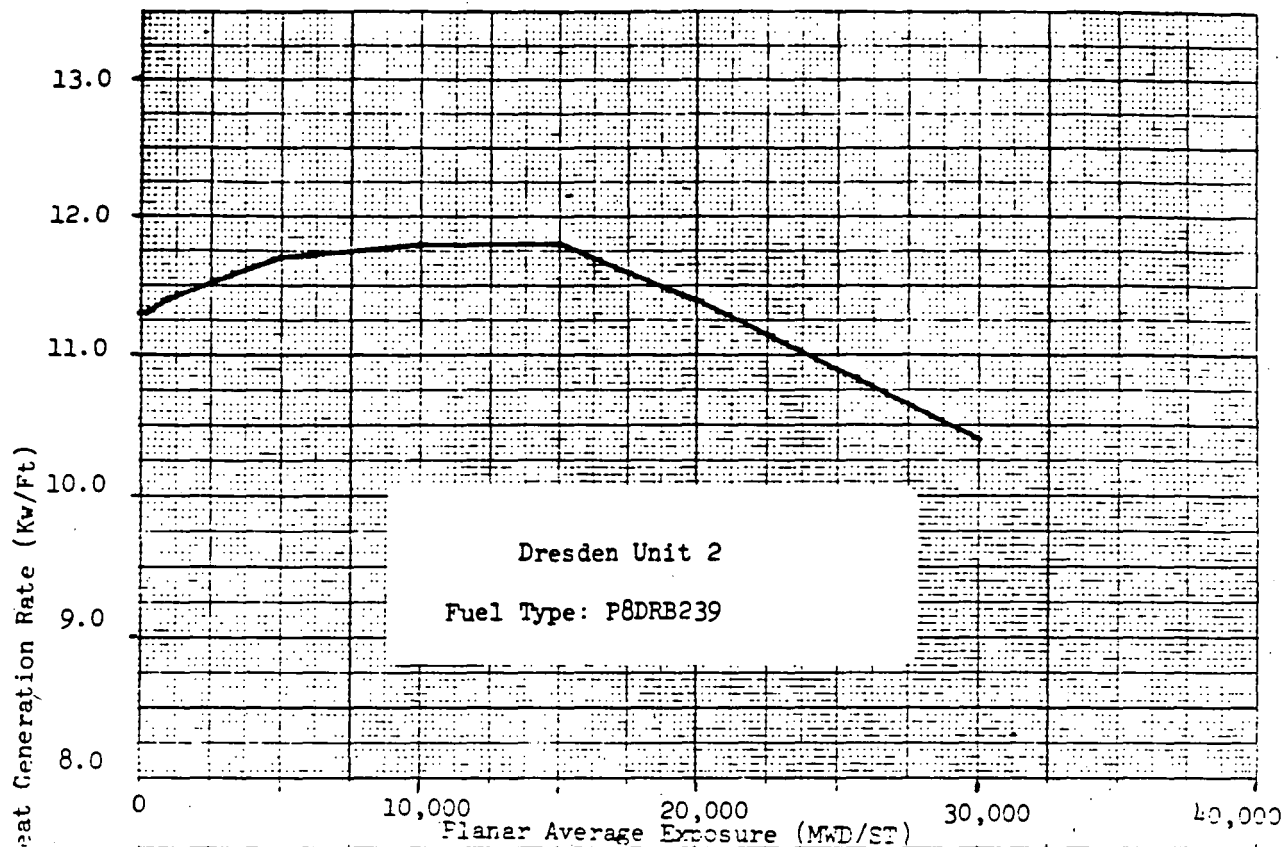MADE IN U.S.A.

NO. 340-M DIETZGEN GRAPH PAPER
MILLIMETER

Maximum Average Planar Linear Heat Generation Rate (Kw/Ft)

Dresden Unit 2

Fuel Type: P8DRB239

Planar Average Exposure (MWD/ST)

Maximum Average Planar Linear Heat Generation Rate (Kw/Ft)

Dresden Unit 2

Fuel Type: P8DRB265H

Planar Average Exposure (MWD/ST)

Figure 3.5-1
(Sheet 2 of 5)

Maximum Average Planar Linear
Heat Generation Rate (MAPLHGR)
vs. Planar Average Exposure

81 C-2
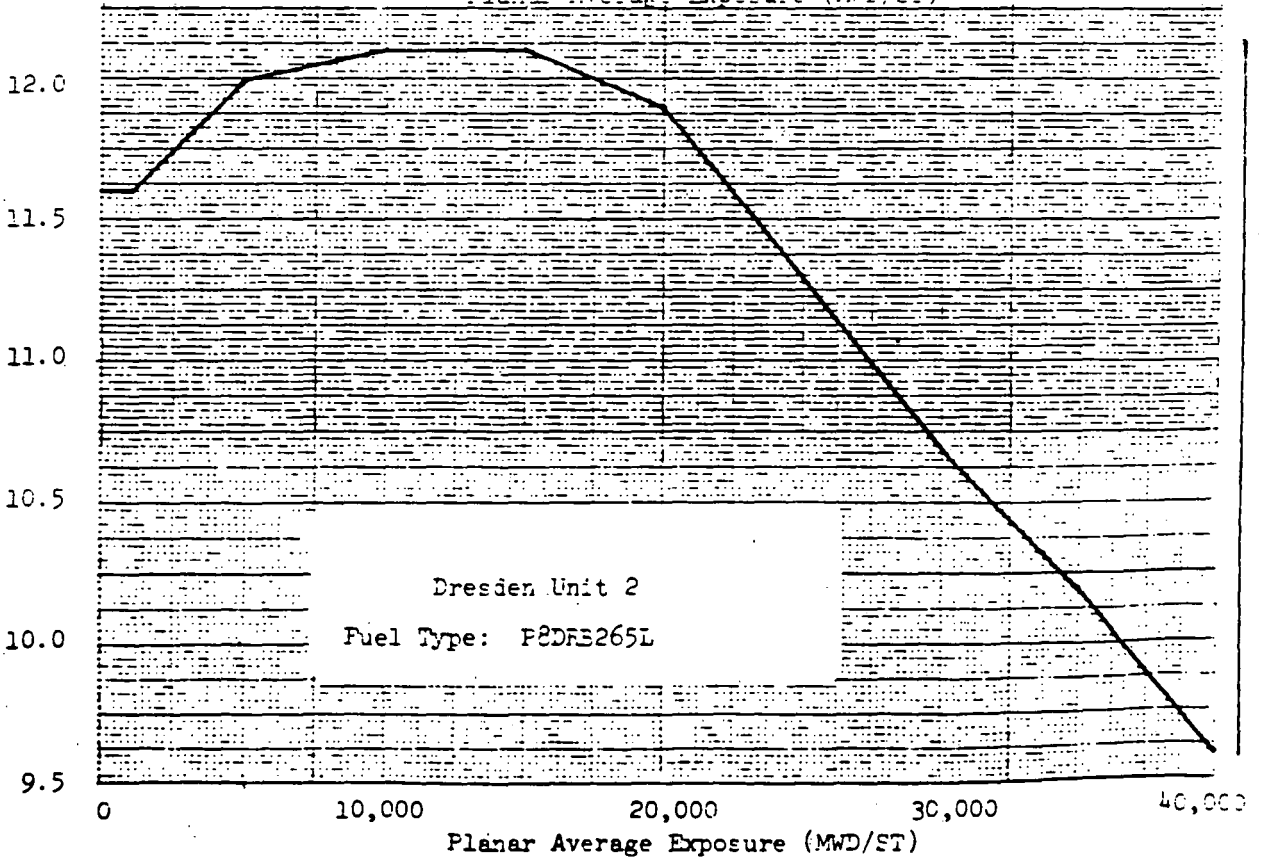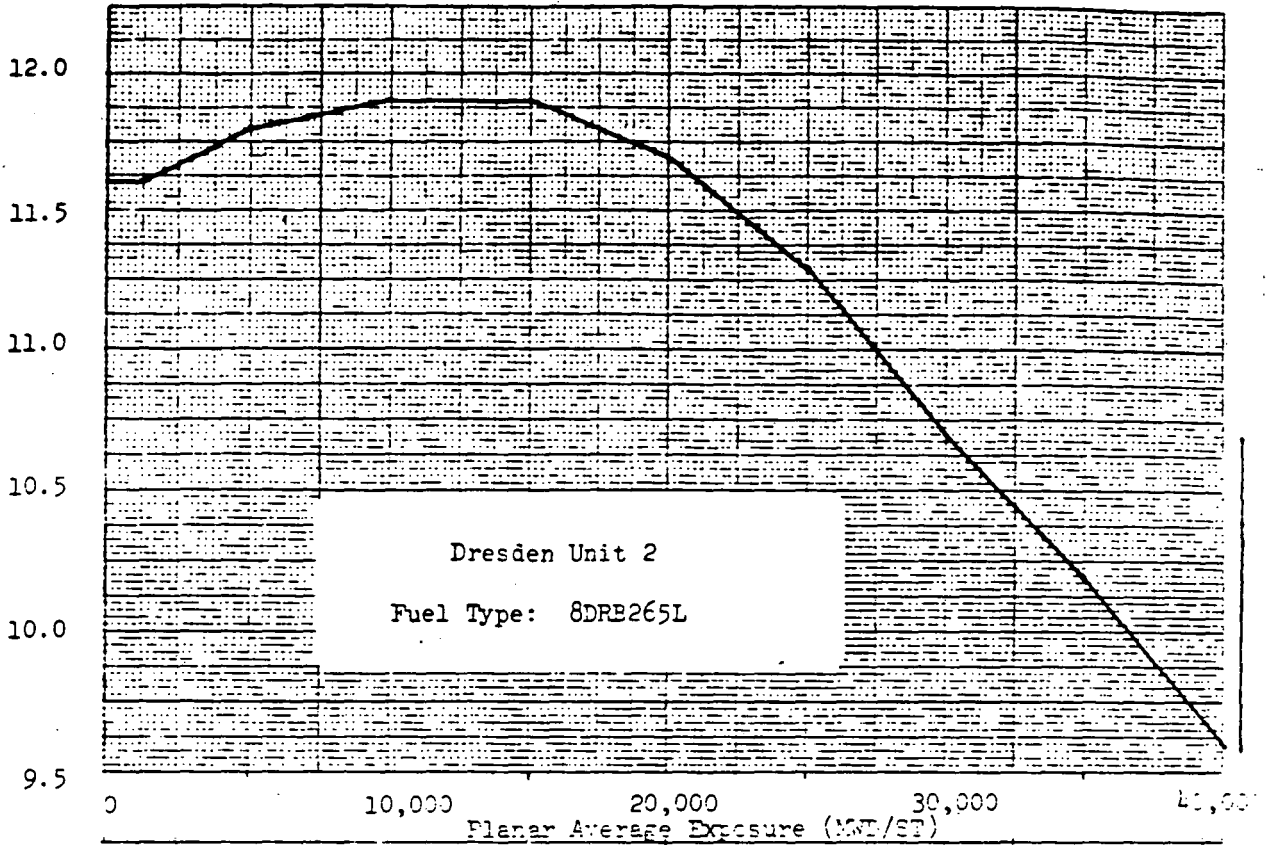
Figure 3.5-1
(Sheet 3 of 5)

Maximum Average Planar Linear
Heat Generation Rate (MAPLHGR)
vs. Planar Average Exposure

81 C-3

Dresden Unit 2

Fuel Type: P8DRB282

Maximum Average Planar Linear Heat Generation Rate (Kw/Ft)

Planar Average Exposure (MWD/ST)

Dresden Unit 2

Fuel Type: XN 8D3.02

Bundle Average Exposure (MWD/MT)

Figure 3.5-1
(Sheet 4 of 5)

Maximum Average Planar Linear Heat Generation Rate (MAPLHGR)
Versus Exposure

81 C-4

Maximum Average Planar Linear Heat Generation Rate (Kw/ft)

Dresden Unit 2

Fuel Type: D2 9x9 2.97 LTA

Bundle Average Exposure (MWD/MT)

10.5

10.0

9.5

9.0

8.5

8.0

10,000   20,000   30,000   40
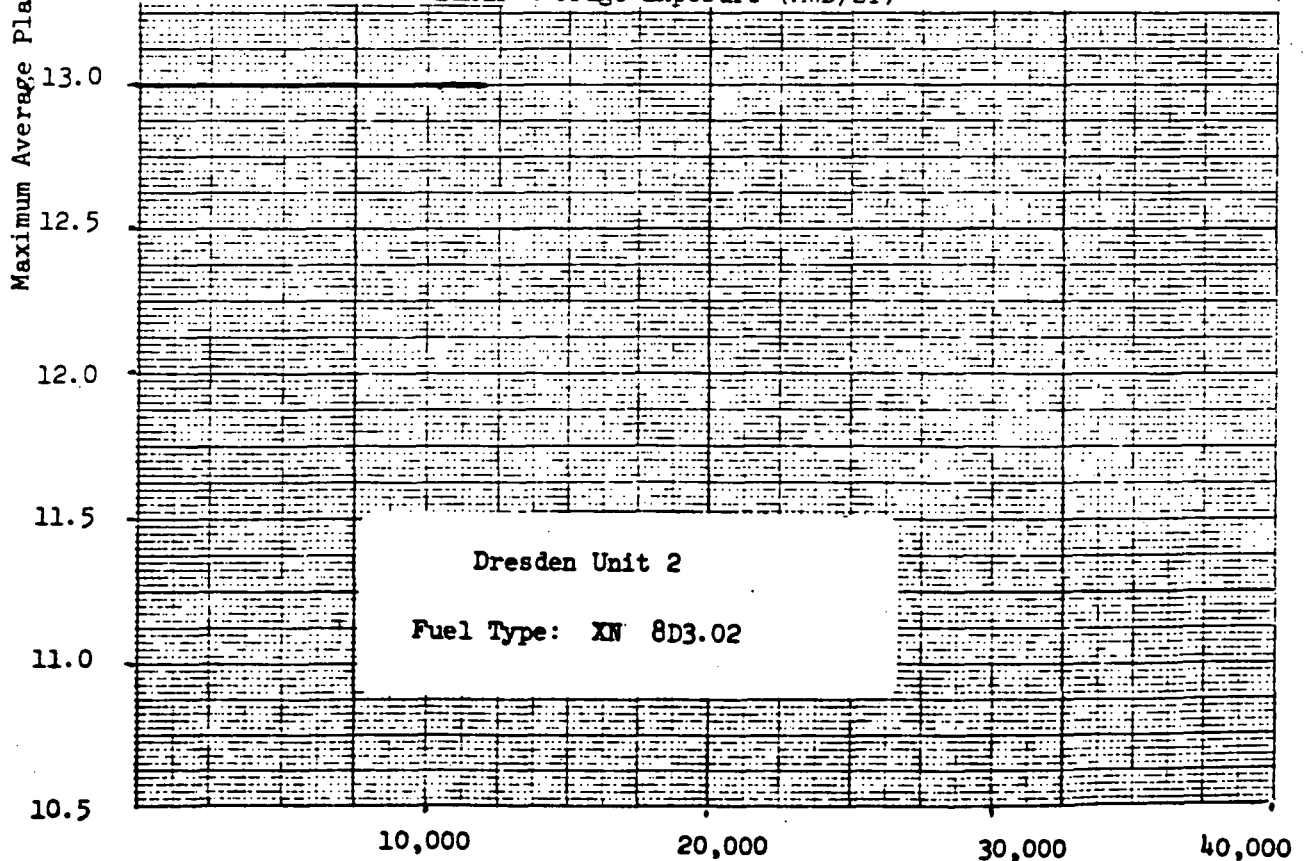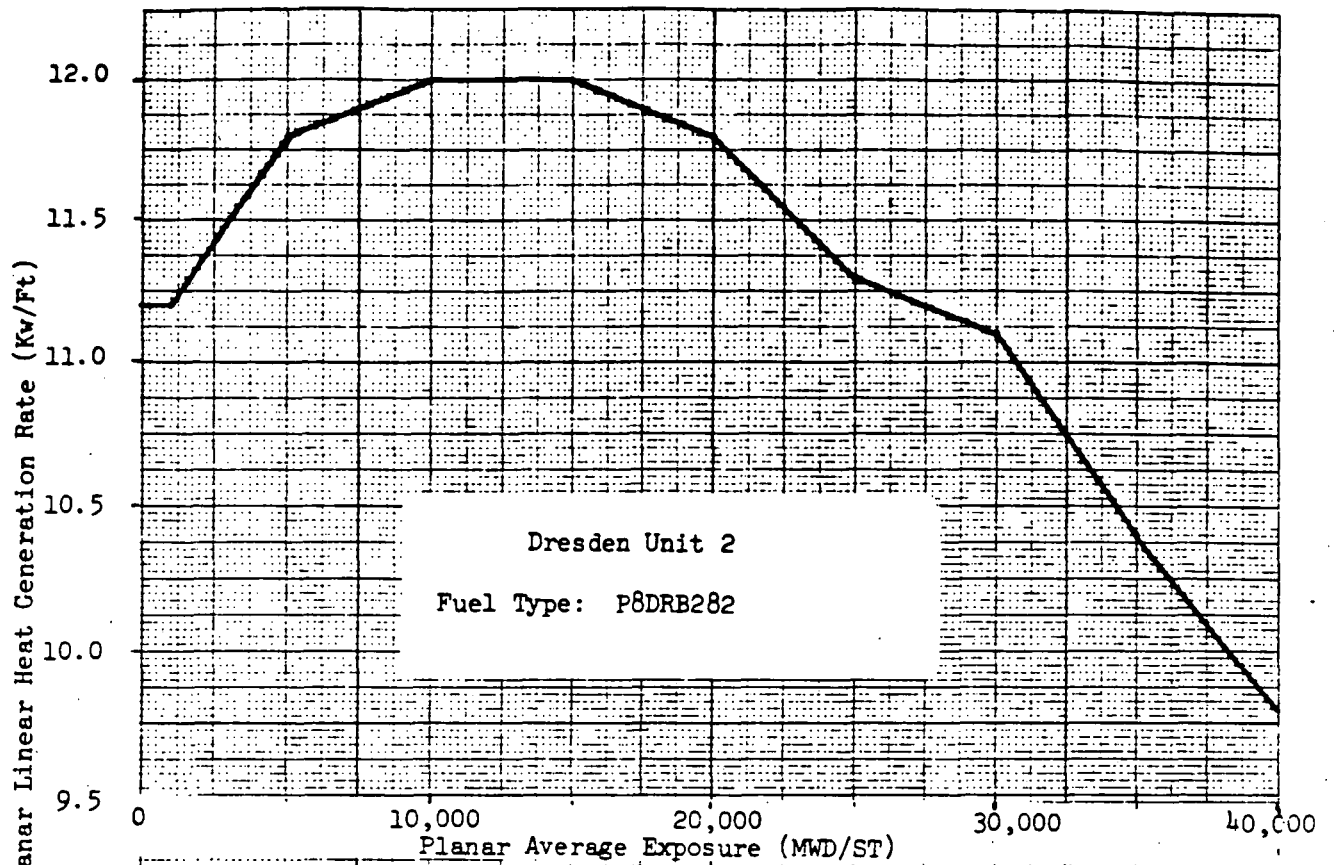
DIETZGEN CORPORATION
MADE IN U.S.A.

NO. 340-M DIETZGEN GRAPH PAPER
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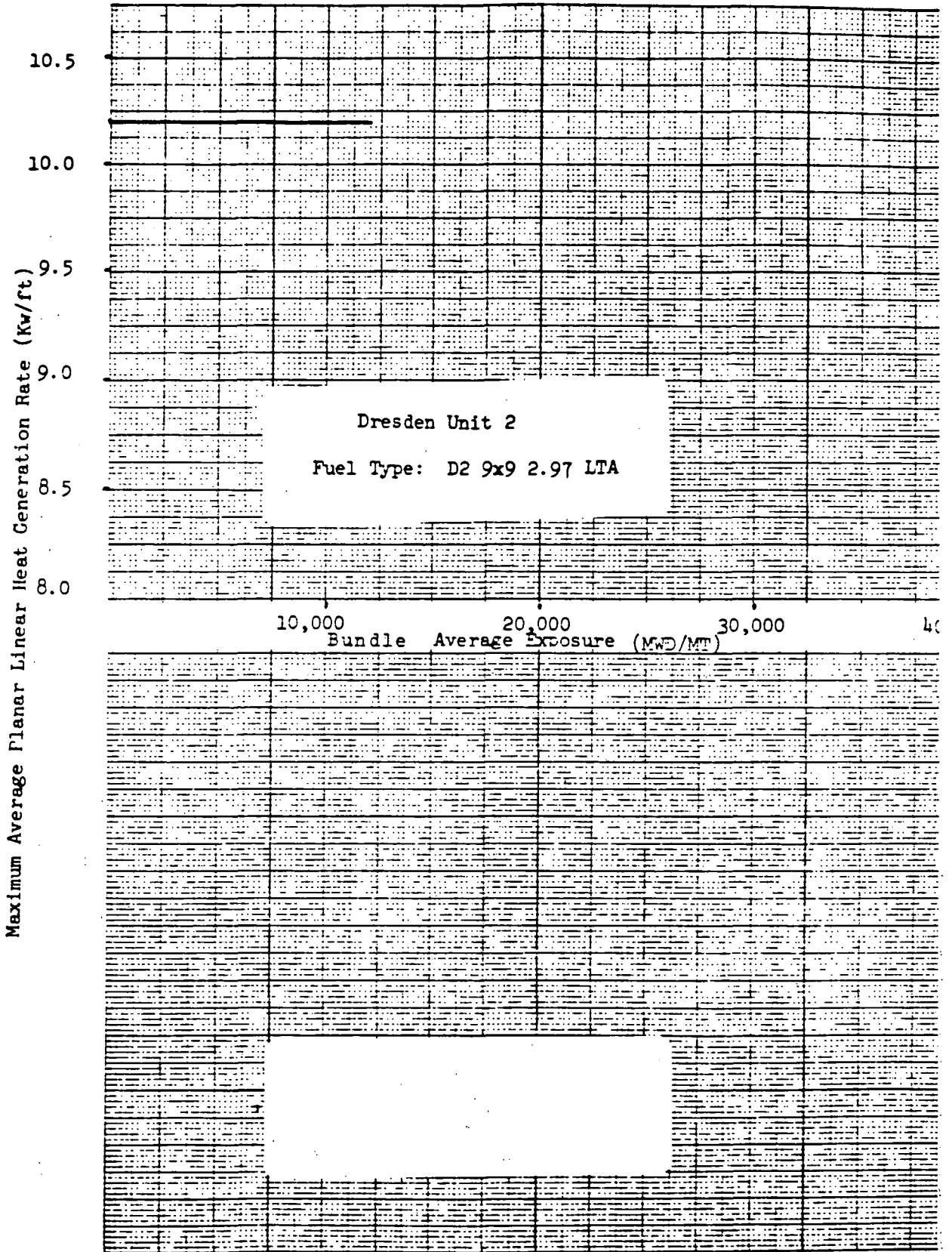
Figure 3.5-1
(Sheet 5 of 5)

Maximum Average Planar Linear
Heat Generation Rate (MAPLHGR)
vs. Bundle Average Exposure

81 C-5

3.5 LIMITING CONDITIONS FOR OPERATION	4.5 SURVEILLANCE REQUIREMENTS

K. Minimum Critical Power Ratio (MCPR)

During steady state operation at rated core flow, MCPR shall be greater than or equal to -
UNIT 2
   1.31 for All 8X8 Fuel Types
   1.35 for XN 9X9 LTA
For core flows other than rated, the MCPR operating limit shall be as follows:

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

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

If at any time during steady state power operation, it is determined that the limiting value for MCPR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the steady state MCPR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

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

K. Minimum Critical Power Ratio (MCPR)

MCPR shall be determined daily during a reactor power operation at $\geq 25\%$ rated thermal power and following any change in power level or distribution that would cause operation with a limiting control rod pattern as described in the bases for Specification 3.3.B.5.
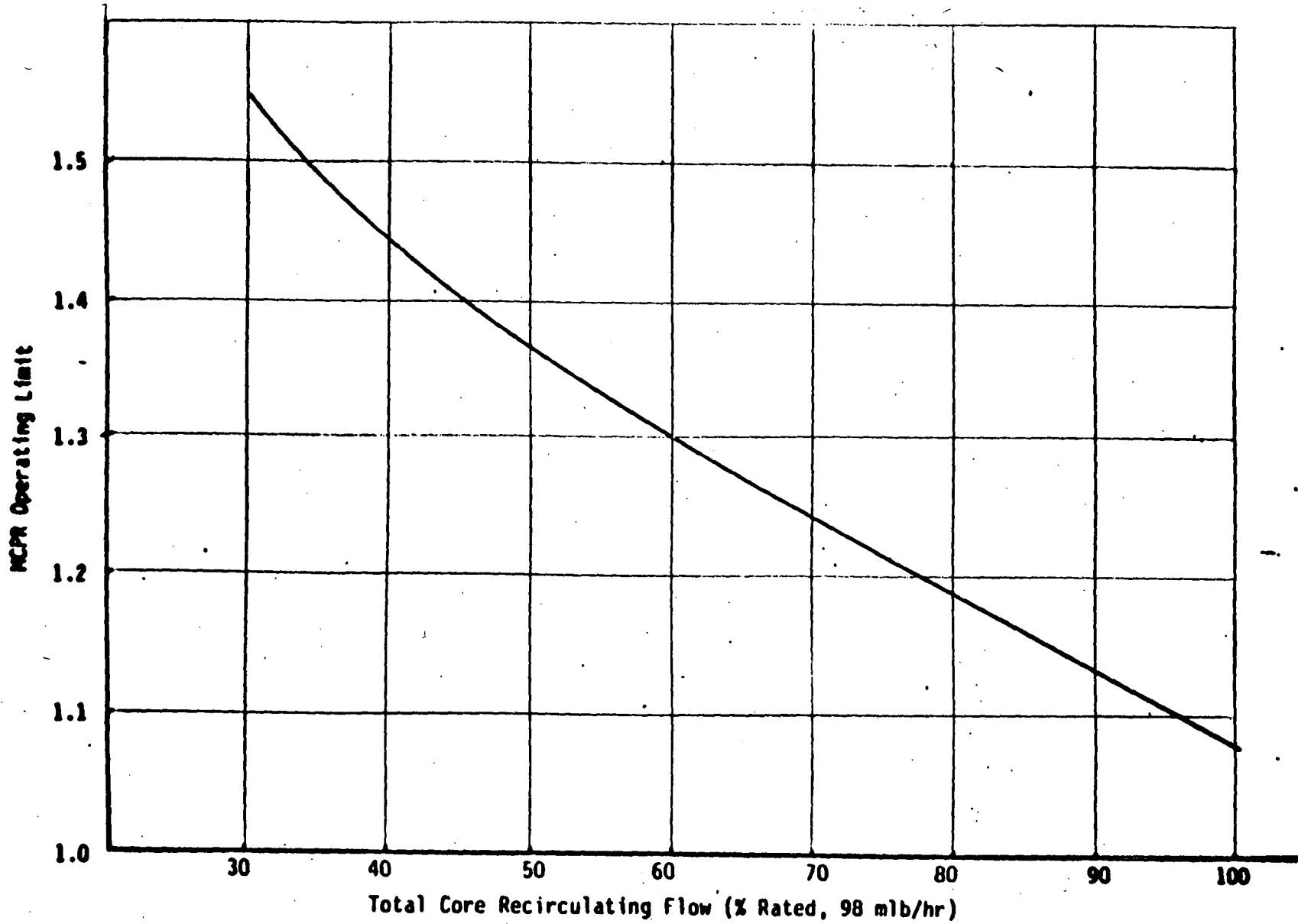
Figure 3.5-2 (Sheet 1 of 2) MCPR Limit for Reduced Core Flow

The chart plots **MCPR Operating Limit** (y-axis, 1.0 to above 1.5) versus **Total Core Recirculating Flow (% Rated, 98 mlb/hr)** (x-axis, 30 to 100).
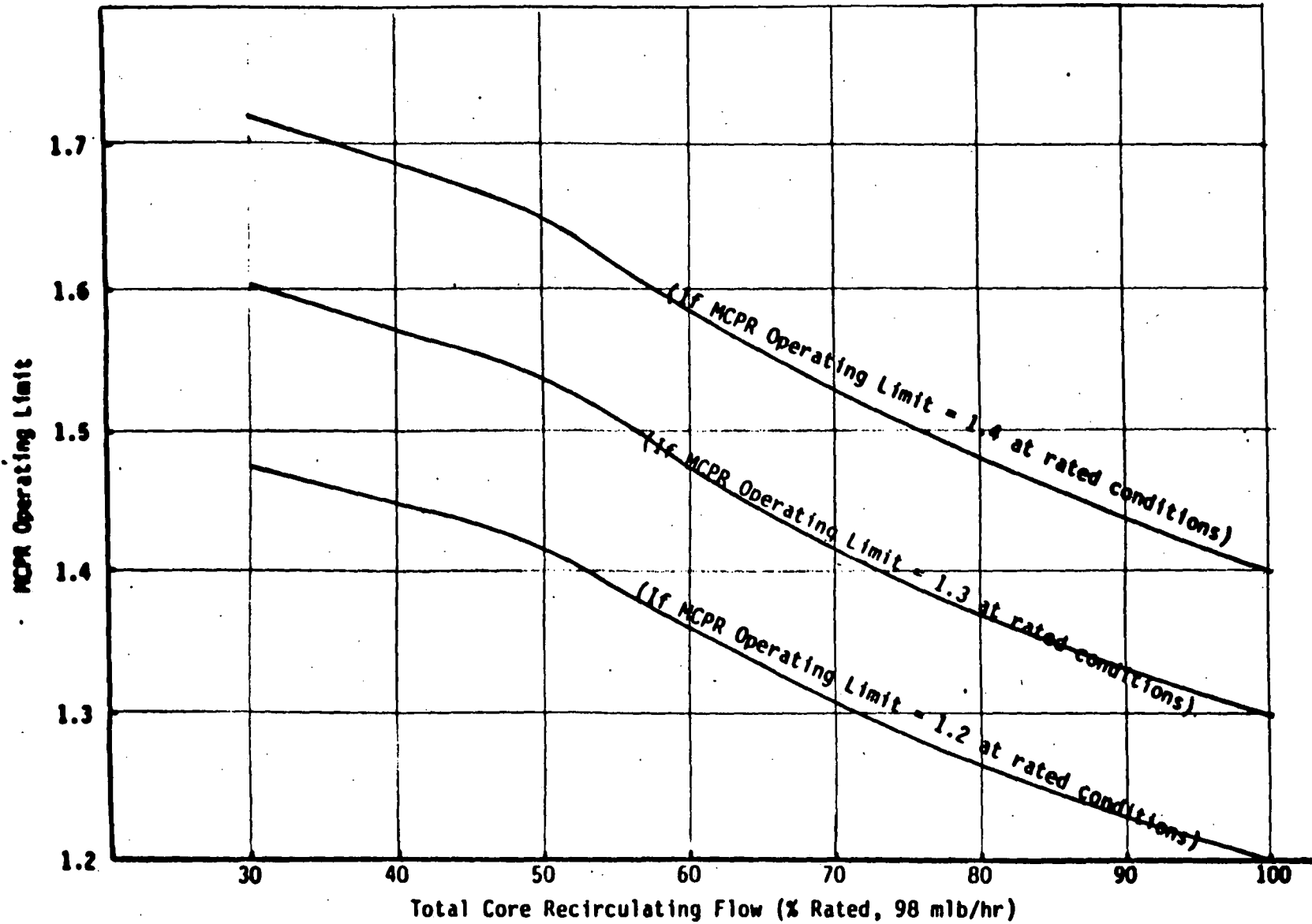
Figure 3.5-2 (Sheet 2 of 2) MCPR Limit for Automatic Flow Control

The chart plots MCPR Operating Limit (y-axis, 1.2 to 1.7) versus Total Core Recirculating Flow (% Rated, 98 mlb/hr) (x-axis, 30 to 100), with three curves labeled:
- (If MCPR Operating Limit = 1.4 at rated conditions)
- (If MCPR Operating Limit = 1.3 at rated conditions)
- (If MCPR Operating Limit = 1.2 at rated conditions)

3.5 Limiting Conditions for Operation Bases

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

Based on the loss of coolant analyses included in References (1), (2) and (4) in accordance with 10CFR50.46 and Appendix K, core cooling systems provide sufficient cooling to the core to dissipate the energy associated with the loss of coolant accident, to limit the calculated peak clad temperature to less than 2200°F, to assure that core geometry remains intact, to limit the core wide clad metal-water reaction to less than 1%, and to limit the calculated local metal-water reaction to less than 17%.

The allowable repair times are established so that the average risk rate for repair would be no greater than the basic risk rate. The method and concept are described in Reference (3). Using the results

developed in this reference, the repair period is found to be less than 1/2 the test interval. This assumes that the core spray and LPCI subsystems constitute a 1 out of 3 system, however, the combined effect of the two systems to limit excessive clad temperatures must also be considered. The test interval specified in Specification 4.5 was 3 months. Therefore, an allowable repair period which maintains the basic risk considering single failures should be less than 45 days and this specification is within this period. For multiple failures, a shorter interval is specified and to improve the assurance that the remaining systems will function, a daily test is called for. Although it is recognized that the information given in reference 3 provides a quantitative method to estimate allowable repair times, the lack of operating data to support the analytical approach prevents complete acceptance of this method at this time. Therefore, the times stated in the specific items were established with due regard to judgment.

Should one core spray subsystem become inoperable, the remaining core spray and the entire LPCI system are available should the

(1) "Loss of Coolant Accident Analyses Report for Dresden Units 2, 3 and Quad-Cities Units 1, 2 Nuclear Power Stations," NEDO-24146A, Revision1, April 1979.

(2) NEDO-20566, General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10CFR50 Appendix K.

(3) APED-"Guidelines for Determining Safe Test Intervals and Repair Times for Engineered Safeguards" - April 1969, I.M. Jacobs and P.W. Marriott.

(4) XN-NF-81-75 "Dresden Unit 3 LOCA Model Using the ENC EXEM Evaluation Model MAPLHGR Results"

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J.5 Limiting Condition for Operation Bases (Cont'd)

I. Average Planar LHGR

This specification assures that the peak cladding temperature following a postulated design basis loss-of-coolant accident will not exceed the 2200°F limit specified in 10CFR50 Appendix K considering the postulated effects of fuel pellet densification.

The peak cladding temperature following a postulated loss-of-coolant accident is primarily a function of the average LHGR of all the rods in a fuel assembly at any axial location and is only dependent secondarily on the rod to rod power distribution within a fuel assembly. Since expected 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 planar LHGR is sufficient to assure that calculated temperatures are below the 10CFR50, Appendix K limit.

The maximum average planar LHGRs shown in Figure 3.5.1 are based on calculations employing the models described in Reference (1) and in reference XN-NF-82-88 Power operation with APLHGRs at or below those shown in Fig. 3.5.1 assures that the peak cladding temperature following a postulated loss-of-coolant accident will not exceed the 2200°F limit.

(1) "Loss of Coolant Accident Analyses Report for Dresden Units 2, 3 and Quad-Cities Units 1, 2 Nuclear Power Stations," NEDO-24146A, Revision 1, April, 1979.

The maximum average planar LHGRs for G.E. fuel plotted in Fig. 3.5.1 at higher exposures result in a calculated peak clad temperature of less than 2200°F. However the maximum average planar LHGRs are shown on Fig. 3.5.1 as limits because conformance calculations have not been performed to justify operation at LHGRs in excess of those shown.

J. Local LHGR

This specification assures that the maximum linear heat generation rate in any fuel rod fabricated by G.E. is less than the design linear

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heat generation rate even if fuel pellet densification is postulated.

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

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

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

the cycle-specific fuel loading, exposure and fuel type. The current cycle's reload licensing analyses identifies the limiting transient for that cycle.

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

For core flows less than rated, the MCPR Operating Limit established in the specification is adjusted to provide protection of the MCPR Safety Limit in the event of an uncontrolled recirculation flow increase to the physical limit of pump flow. This protection is provided for manual and automatic flow control by choosing the MCPR operating limit as the value from Figure 3.5-2 Sheet 1 or the rated core flow value, whichever is greater. For Automatic Flow Control, in addition to protecting the MCPR Safety Limit during the flow run-up event, protection is provided against violating the rated flow MCPR Operating Limit during an automatic flow increase to rated core flow. This protection is provided by the reduced flow MCPR limits shown in Figure 3.5-2 Sheet 2 where the curve corresponding to the current rated flow MCPR limit is used (linear interpolation between the MCPR limit lines depicted is permissible).

Therefore, for Automatic Flow Control, the MCPR Operating Limit is chosen as the value from Figure 3.5-2 Sheet 1, Sheet 2 or the rated flow value, whichever is greatest. It should be noted that if the rated flow MCPR Limit must be increased due to degradation of control rod scram times during the current cycle, the new value of the rated flow MCPR limit is applied when using Figure 3.5-2 Sheet 2.

I. **Average Planar LHGR**

At core thermal power levels less than or
equal to 25 per cent, operating plant
experience and thermal hydraulic analyses
indicate that the resulting average planar
LHGR is below the maximum average planar LHGR
by a considerable margin; therefore, evaluation
of the average planar LHGR below this power
level is not necessary. The daily require-
ment for calculating average planar LHGR
above 25 per cent rated thermal power is
sufficient since power distribution shifts
are slow when there have not been signifi-
cant power or control rod changes.

J. **Local LHGR**

The LHGR for G.E. fuel shall
be checked daily during reactor operation at
greater than or equal to 25 per cent power to
determine if fuel burnup or control rod movement
has caused changes in power distribution.
A limiting LHGR value is precluded by a
considerable margin when employing a per
missible control rod pattern below 25% rated
thermal power.

K. **Minimum Critical Power Ratio (MCPR)**

At core thermal power levels less than or equal
to 25 per cent, the reactor will be operating
at minimum recirculation pump speed and the
moderator void content will be very small. For
all designated control rod patterns which may be
employed at this point, operating plant experience
and thermal hydraulic analysis indicates that the
resulting MCPR value is in excess of requirements
by a considerable margin. With this low void
content, any inadvertent core flow increase
would only place operation in a more con-
servative mode relative to MCPR.

The daily requirement for calculating
MCPR above 25 percent rated thermal
power is sufficient since power distribution
shifts are very slow when there have not been
significant power or control rod changes.

In addition, the K_f correction applied to
the LCO provides margin for flow increase
from low flows.

NUCLEAR ENERGY BUSINESS GROUP • GENERAL ELECTRIC COMPANY
SAN JOSE, CALIFORNIA 95125

GENERAL ⚡ ELECTRIC

APPLICABLE TO:

PUBLICATION NO. NEDO-24146A

T. I. E. NO. 79NED273

TITLE LOCA ANALYSIS REPORT FOR
DRESDEN UNITS 2, 3 AND QUAD
CITIES 1,2 NUCLEAR POWER STATIONS

ISSUE DATE April 1979

ERRATA And ADDENDA SHEET

NO. 7

DATE July 1982

NOTE: *Correct all copies of the applicable publication as specified below.*

ITEM	REFERENCES (SECTION, PAGE PARAGRAPH, LINE)	INSTRUCTIONS (CORRECTIONS AND ADDITIONS)
1.	Page 4-11	Replace with new page 4-11.
2.	Page 4-14	Replace with new page 4-14.

Table 4I

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

PLANT: Dresden 2,3/Quad Cities 1,2   FUEL TYPE: P8DRB282

Average Planar Exposure (MWd/t)	MAPLHGR (kW/ft)	PCT (°F)	Oxidation Fraction
200	11.2	2132	0.029
1,000	11.2	2132	0.029
5,000	11.8	2183	0.033
10,000	12.0	2189	0.032
15,000	12.0	2199	0.033
20,000	11.8	2181	0.032
25,000	11.3	2110	0.025
30,000	11.1	2075	0.043
35,000	10.4	1981	0.035
40,000	9.8	1886	0.021

NOTE: Credit taken for the effects of pre-pressurization of the fuel rods

Table 4J

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

PLANT: Dresden 2,3/Quad Cities 1,2   FUEL TYPE: P8DRB265H

Average Planar Exposure (MWd/t)	MAPLHGR (kW/ft)	PCT (°F)	Oxidation Fraction
200	11.5	2163	0.032
1,000	11.6	2171	0.032
5,000	11.9	2192	0.033
10,000	12.1	2198	0.033
15,000	12.1	2200	0.033
20,000	11.9	2190	0.032
25,000	11.3	2116	0.026
30,000	10.7	2018	0.018
35,000	10.2	1934	0.022
40,000	9.6	1835	0.009

NOTE: Same as Table 4I

Table 4K

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

PLANT: Dresden 2,3/Quad Cities 1,2   FUEL TYPE: P8DRB239

Average Planar Exposure (MWd/t)	MAPLHGR (kW/ft)	PCT (°F)	Oxidation Fraction
200	11.3	2117	0.027
1,000	11.4	2129	0.028
5,000	11.7	2160	0.030
10,000	11.8	2141	0.028
15,000	11.8	2145	0.028
20,000	11.4	2114	0.025
25,000	10.9	2041	0.020
30,000	10.4	1968	0.015

NOTE: Same as Table 4I

Table 4L

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

PLANT: Quad Cities 2   FUEL TYPE: P8DGB284 (Barrier Fuel)

Average Planar Exposure (MWd/t)	MAPLHGR (kW/ft)	PCT (°F)	Oxidation Fraction
200	11.5	2152	0.031
1,000	11.5	2150	0.030
5,000	11.9	2185	0.033
10,000	12.0	2189	0.032
15,000	12.1	2199	0.033
20,000	11.9	2180	0.032
25,000	11.3	2109	0.025
30,000	10.9	2042	0.020
35,000	10.3	1956	0.030
40,000	9.7	1846	0.010

NOTE: Same as Table 4I

Table 4M

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

PLANT: Quad Cities 2   FUEL TYPE: P8DGB263L (Barrier Fuel)

Average Planar Exposure (MWd/t)	MAPLHGR (kW/ft)	PCT (°F)	Oxidation Fraction
200	10.2	1991	0.018
1,000	10.3	2003	0.018
5,000	11.5	2133	0.027
10,000	12.0	2185	0.032
15,000	12.1	2199	0.033
20,000	11.9	2187	0.032
25,000	11.4	2116	0.026
30,000	10.8	2020	0.019
35,000	10.2	1919	0.013
40,000	9.5	1829	0.009

NOTE: Same as Table 4I

Table 4N

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

PLANT: Quad Cities 2   FUEL TYPE: P8DGB263H (Barrier Fuel)

Average Planar Exposure (MWd/t)	MAPLHGR (kW/ft)	PCT (°F)	Oxidation Fraction
200	10.1	1978	0.017
1,000	10.2	1982	0.017
5,000	11.2	2102	0.025
10,000	12.1	2196	0.033
15,000	12.1	2198	0.033
20,000	11.9	2188	0.032
25,000	11.3	2116	0.026
30,000	10.8	2022	0.019
35,000	10.2	1921	0.018
40,000	9.5	1830	0.009

NOTE: Same as Table 4I

Table 40
MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

PLANT: Quad Cities 2   FUEL TYPE: P8DGB298 (Barrier Fuel)

Average Planar Exposure (MWd/t)	MAPLHGR (kW/ft)	PCT (°F)	Oxidation Fraction
200	9.9	1962	0.016
1,000	9.9	1959	0.016
5,000	10.4	2000	0.018
10,000	11.7	2148	0.028
15,000	12.0	2198	0.033
20,000	11.9	2194	0.033
25,000	11.6	2153	0.029
30,000	10.9	2074	0.034
35,000	10.2	1963	0.032
40,000	9.5	1827	0.009

NOTE: Same as Table 4I.

Table 4P
MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

PLANT: Quad Cities 1,2   FUEL TYPES: P8DRB265L, P8DGB265L (Barrier Fuel)

Average Planar Exposure (MWd/t)	MAPLHGR (kW/ft)	PCT (°F)	Oxidation Fraction
200	11.6	2171	0.032
1,000	11.6	2177	0.033
5,000	12.0	2198	0.034
10,000	12.1	2195	0.033
15,000	12.1	2199	0.033
20,000	11.9	2187	0.032
25,000	11.3	2115	0.025
30,000	10.7	2018	0.018
35,000	10.2	1927	0.022
40,000	9.6	1834	0.009

NOTE: Same as Table 4I.