ENCLOSURE 2

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNITS 1, 2, AND 3

PROPOSED TECHNICAL SPECIFICATION (TS) CHANGE TS-377
MARKED PAGES

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II. **MARKED PAGES**

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1.1/2.1 FUEL CLADDING INTEGRITY

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1.1 FUEL CLADDING INTEGRITY	**2.1 FUEL CLADDING INTEGRITY**

SAFETY LIMIT

1.1 FUEL CLADDING INTEGRITY

Applicability

Applies to the interrelated variables associated with fuel thermal behavior.

Objective

To establish limits which ensure the integrity of the fuel cladding.

Specifications

A. Thermal Power Limits

1. Reactor Pressure >800 psia and Core Flow > 10% of Rated.

   When the reactor pressure is greater than 800 psia, the existence of a minimum critical power ratio (MCPR) less than 1.07 1.10 shall constitute violation of the fuel cladding integrity safety limit.

LIMITING SAFETY SYSTEM SETTING

2.1 FUEL CLADDING INTEGRITY

Applicability

Applies to trip settings of the instruments and devices which are provided to prevent the reactor system safety limits from being exceeded.

Objective

To define the level of the process variables at which automatic protective action is initiated to prevent the fuel cladding integrity safety limit from being exceeded.

Specifications

The limiting safety system settings shall be as specified below:

A. Neutron Flux Trip Settings

1. APRM Flux Scram Trip Setting (Run Mode) (Flow biased)

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

1.1   BASES: FUEL CLADDING INTEGRITY SAFETY LIMIT

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

The fuel cladding integrity limit is set such that no calculated
fuel damage would occur as a result of an abnormal operational
transient. Because fuel damage is not directly observable, the Fuel
Cladding Safety Limit is defined with margin to the conditions which
would produce onset transition boiling (MCPR of 1.0). ~~This
establishes a Safety Limit such that the minimum critical power
ratio (MCPR) is no less than 1.07.~~ MCPR > 1.07 represents a
conservative margin relative to the conditions required to maintain
fuel cladding integrity.

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. Since boiling transition is not a
directly observable parameter, 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, i.e., normal plant operation
presented on Figure 2.1-1 by the nominal expected flow control
line. The Safety Limit ~~(MCPR of 1.07)~~ has sufficient conservatism
to assure that in the event of an abnormal operational transient
initiated from a normal operating condition (MCPR > limits specified
in Specification 3.5.K) more than 99.9 percent of the fuel rods in
the core are expected to avoid boiling transition. The margin
between MCPR of 1.0 (onset of transition boiling) and the safety
limit ~~1.07~~ is derived from a detailed statistical analysis
considering all of the uncertainties in monitoring the core
operating state including uncertainty in the boiling transition
correlation as described in Reference 1. The uncertainties employed
in deriving the safety limit are provided at the beginning of each
fuel cycle.

MCPR

Maintaining the MCPR greater than the Safety Limit MCPR

1.1 BASES (Cont'd)

Because the boiling transition correlation is based on a large quantity of full scale data there is a very high confidence that operation of a fuel assembly at the condition of MCPR ≥ 1.07 would not produce boiling transition. Thus, although it is not required to establish the safety limit additional margin exists between the safety limit and the actual occurrence of loss of cladding integrity.

However, if boiling transition were to occur, clad perforation would not be expected. Cladding temperatures would increase to approximately 1100°F which is below the perforation temperature of the cladding material. This has been verified by tests in the General Electric Test Reactor (GETR) where fuel similar in design to BFNP operated above the critical heat flux for a significant period of time (30 minutes) without clad perforation.

If reactor pressure should ever exceed 1400 psia during normal power operation (the limit of applicability of the boiling transition correlation) it would be assumed that the fuel cladding integrity Safety Limit has been violated.

At pressures below 800 psia, the core elevation pressure drop (0 power, 0 flow) is greater than 4.56 psi. At low powers and flows this pressure differential is maintained in the bypass region of the core. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low powers and flow will always be greater than 4.56 psi. Analyses show that with a flow of 28×10^3 lbs/hr bundle flow, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.56 psi driving head will be greater than 28×10^3 lbs/hr. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors this corresponds to a core thermal power of more than 50 percent. Thus, a core thermal power limit of 25 percent for reactor pressures below 800 psia is conservative.

For the fuel in the core 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 water level should drop below the top of the fuel during this time, the ability to remove decay heat is reduced. This reduction in cooling capability could lead to elevated cladding temperatures and clad perforation. As long as the fuel remains covered with water, sufficient cooling is available to prevent fuel clad perforation.

equal to the Safety Limit MCPR

2.1 <u>BASES</u> (Cont'd)

Analyses of the limiting transients show that no scram adjustment is required to assure MCPR ≥ 1.07 when the transient is initiated from MCPR limits specified in Specification 3.5.k.

2. <u>APRM Flux Scram Trip Setting (REFUEL or STARTUP/HOT STANDBY MODE)</u>

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 setpoint and the safety limit, 25 percent of rated. The margin is adequate to accommodate anticipated maneuvers associated with power plant startup. Effects of increasing pressure at zero or low void content are minor, cold water from sources available during startup is not much colder than that already in the system, temperature coefficients are small, and control rod patterns are constrained to be uniform by operating procedures backed up by the rod worth minimizer. Thus, of all possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant 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 approach 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 to assure a scram before the power could exceed the safety limit. The 15 percent APRM scram remains active until the mode switch is placed in the RUN position. This switch occurs when reactor pressure is greater than 850 psig.

3. <u>IRM Flux Scram Trip Setting</u>

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 covered by the IRM by means of a range switch and the 5 decades are broken down into 10 ranges, each being one-half of a decade in size. The IRM scram 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 at 120 divisions for that range; likewise if the instrument was on range 5, the scram setting would be 120 divisions on that range.

is greater than the Safety Limit MCPR

2.1 <u>BASES</u> (Cont'd)

<u>IRM Flux Scram Trip Setting (Continued)</u>

Thus, as the IRM is ranged up to accommodate the increase in power level, the scram setting is also ranged up. A scram at 120 divisions on the IRM instruments remains in effect as long as the reactor is in the startup mode. In addition, the APRM 15 percent scram prevents higher power operation without being in the RUN mode. The IRM scram provides protection for changes which occur both locally and over the entire core. The most significant sources of reactivity change during the power increase are due to control rod withdrawal. For insequence control rod withdrawal, the rate of change of power is slow enough due to the physical limitation of withdrawing control rods that heat flux is in equilibrium with the neutron flux. An IRM scram would result in a reactor shutdown well before any SAFETY LIMIT is exceeded. For the case of a single control 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. This condition exists at quarter rod density. Quarter rod density is illustrated in paragraph 7.5.5 of the FSAR. 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 1.07. Based on the above analysis, the IRM provides protection against local control rod withdrawal errors and continuous withdrawal of control rods in sequence.

the Safety Limit MCPR

4. <u>Fixed High Neutron Flux Scram Trip</u>

The average power range monitoring (APRM) system, which is calibrated using heat balance data taken during steady-state conditions, reads in percent of rated power (3,293 MWt). The APRM system responds directly to neutron flux. Licensing analyses have demonstrated that with a neutron flux scram of 120 percent of rated power, none of the abnormal operational transients analyzed violate the fuel SAFETY LIMIT and there is a substantial margin from fuel damage.

B.   <u>APRM Control Rod Block</u>

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 rod withdrawal beyond a given point at constant recirculation flow rate and thus prevents scram actuation. This rod block trip setting, which is automatically varied with recirculation loop flow rate, prevents an increase in the reactor power level to excess values due to control rod withdrawal. The flow variable trip setting is selected to provide adequate margin to the flow-biased scram setpoint.

2.1 BASES (Cont'd)

C. Reactor Water Low Level Scram and Isolation (Except Main Steam Lines)

The setpoint for the low level scram is above the bottom of the
separator skirt. This level has been used in transient analyses
dealing with coolant inventory decrease. The results reported in
FSAR subsection 14.5 show that scram and isolation of all process
lines (except main steam) at this level adequately protects the fuel
and the pressure barrier, because MCPR is greater than 1.07, in all
cases, and system pressure does not reach the safety valve
settings. The scram setting is sufficiently below normal operating
range to avoid spurious scrams.

                                              the Safety Limit MCPR

D. Turbine Stop Valve Closure Scram

The turbine stop valve closure trip anticipates the pressure,
neutron flux and heat flux increases that would result from closure
of the stop valves. With a trip setting of 10 percent of valve
closure from full open, the resultant increase in heat flux is such
that adequate thermal margins are maintained even during the worst
case transient that assumes the turbine bypass valves remain
closed. (Reference 2)

E. Turbine Control Valve Fast Closure or Turbine Trip Scram

Turbine control valve fast closure or turbine trip scram anticipates
the pressure, neutron flux, and heat flux increase that could result
from control valve fast closure due to load rejection or control
valve closure due to turbine trip: each without bypass valve
capability. The reactor protection system initiates a scram in less
than 30 milliseconds after the start of control valve fast closure
due to load rejection or control valve closure due to turbine trip.
This scram is achieved by rapidly reducing hydraulic control oil
pressure at the main turbine control valve actuator disc dump
valves. This loss of pressure is sensed by pressure switches whose
contacts form the one-out-of-two-twice logic input to the reactor
protection system. This trip setting, a nominally 50 percent
greater closure time and a different valve characteristic from that
of the turbine stop valve, combine to produce transients very
similar to that for the stop valve. No significant change in MCPR
occurs. Relevant transient analyses are discussed in References 2
and 3 of the Final Safety Analysis Report. This scram is bypassed
when turbine steam flow is below 30 percent of rated, as measured by
turbine first state pressure.

3.3/4.3   BASES (Cont'd)

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 RBM channels are provided, and one of these may be bypassed from the console for maintenance and/or testing. Automatic rod withdrawal blocks from one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage. 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.

C.   Scram Insertion Times

The control rod system is designated to bring the reactor subcritical at the rate fast enough to prevent fuel damage; i.e., to prevent the MCPR from becoming less than 1.07. The limiting power transient is given in Reference 1. Analysis of this transient shows that the negative reactivity rates resulting from the scram with the average response of all the drives as given in the above specification provide the required protection, and MCPR remains greater than 1.07.

On an early BWR, some degradation of control rod scram performance occurred during plant startup and was determined to be caused by particulate material (probably construction debris) plugging an internal control rod drive filter. The design of the present control rod drive (Model 7RDB144B) is grossly improved by the relocation of the filter to a location out of the scram drive path; i.e., it can no longer interfere with scram performance, even if completely blocked.

The degraded performance of the original drive (CRD7RDB144A) under dirty operating conditions and the insensitivity of the redesigned drive (CRD7RDB144B) has been demonstrated by a series of engineering tests under simulated reactor operating conditions. The successful performance of the new drive under actual operating conditions has also been demonstrated by consistently good in-service test results for plants using the new drive and may be inferred from plants using the older model

the Safety Limit MCPR

1.1/2.1 FUEL CLADDING INTEGRITY

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING

1.1 FUEL CLADDING INTEGRITY

Applicability

Applies to the interrelated variables associated with fuel thermal behavior.

Objective

To establish limits which ensure the integrity of the fuel cladding.

Specifications

A. Thermal Power Limits

    1. Reactor Pressure >800 psia and Core Flow > 10% of Rated.

       When the reactor pressure is greater than 800 psia, the existence of a minimum critical power ratio (MCPR) less than 1.07 ~~1.07~~ 1.10 shall constitute violation of the fuel cladding integrity safety limit.

2.1 FUEL CLADDING INTEGRITY

Applicability

Applies to trip settings of the instruments and devices which are provided to prevent the reactor system safety limits from being exceeded.

Objective

To define the level of the process variables at which automatic protective action is initiated to prevent the fuel cladding integrity safety limit from being exceeded.

Specifications

The limiting safety system settings shall be as specified below:

A. Neutron Flux Trip Settings

    1. APRM Flux Scram Trip Setting (RUN Mode) (Flow Biased)

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

1.1 BASES: FUEL CLADDING INTEGRITY SAFETY LIMIT

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

The fuel cladding integrity limit is set such that no calculated fuel damage would occur as a result of an abnormal operational transient. Because fuel damage is not directly observable, the Fuel Cladding Safety Limit is defined with margin to the conditions which would produce onset transition boiling (MCPR of 1.0). ~~This establishes a Safety Limit such that the minimum critical power ratio (MCPR) is no less than 1.07.~~ MCPR > 1.07, represents a conservative margin relative to the conditions required to maintain fuel cladding integrity.

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. Since boiling transition is not a directly observable parameter, 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, i.e., normal plant operation presented on Figure 2.1-1 by the nominal expected flow control line. The Safety Limit ~~(MCPR of 1.07)~~ has sufficient conservatism to assure that in the event of an abnormal operational transient initiated from a normal operating condition (MCPR > limits specified in Specification 3.5.K) more than 99.9 percent of the fuel rods in the core are expected to avoid boiling transition. The margin between MCPR of 1.0 (onset of transition boiling) and the safety limit ~~1.07~~ is derived from a detailed statistical analysis considering all of the uncertainties in monitoring the core operating state including uncertainty in the boiling transition correlation as described in Reference 1. The uncertainties employed in deriving the safety limit are provided at the beginning of each fuel cycle.

MCPR

Maintaining the MCPR greater than the Safety Limit MCPR

BFN
Unit 2

1.1/2.1-8

TS 370
Letter Dated   95

Because the boiling transition correlation is based on a large quantity of full scale data there is a very high confidence that operation of a fuel assembly at the condition of MCPR ≠ 1.07 would not produce boiling transition. Thus, although it is not required to establish the safety limit additional margin exists between the safety limit and the actual occurrence of loss of cladding integrity.

However, if boiling transition were to occur, clad perforation would not be expected. Cladding temperatures would increase to approximately 1,100°F which is below the perforation temperature of the cladding material. This has been verified by tests in the General Electric Test Reactor (GETR) where fuel similar in design to BFNP operated above the critical heat flux for a significant period of time (30 minutes) without clad perforation.

If reactor pressure should ever exceed 1,400 psia during normal power operation (the limit of applicability of the boiling transition correlation) it would be assumed that the fuel cladding integrity Safety Limit has been violated.

At pressures below 800 psia, the core elevation pressure drop (0 power, 0 flow) is greater than 4.56 psi. At low powers and flows this pressure differential is maintained in the bypass region of the core. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be greater than 4.5 psi. Analyses show that with a flow of 28×10^3 lbs/hr bundle flow, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.56 psi driving head will be greater than 28×10^3 lbs/hr. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors this corresponds to a core thermal power of more than 50 percent. Thus, a core thermal power limit of 25 percent for reactor pressures below 800 psia is conservative.

For the fuel in the core 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 water level should drop below the top of the fuel during this time, the ability to remove decay heat is reduced. This reduction in cooling capability could lead to elevated cladding temperatures and clad perforation. As long as the fuel remains covered with water, sufficient cooling is available to prevent fuel clad perforation.

equal to the Safety Limit MCPR

2.1 BASES (Cont'd)

Analyses of the limiting transients show that no scram adjustment is required to assure MCPR ≥ 1.07 when the transient is initiated from MCPR limits specified in Specification 3.5.k.

2. APRM Flux Scram Trip Setting (REFUEL or STARTUP/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 setpoint and the safety limit, 25 percent of rated. The margin is adequate to accommodate anticipated maneuvers associated with power plant startup. Effects of increasing pressure at zero or low void content are minor, cold water from sources available during startup is not much colder than that already in the system, temperature coefficients are small, and control rod patterns are constrained to be uniform by operating procedures backed up by the rod worth minimizer. Worth of individual rods is very low in a uniform rod pattern. Thus, of all possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant 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 approach to the scram level, the rate of power rise is no more than five percent of rated power per minute, and the APRM system would be more than adequate to assure a scram before the power could exceed the safety limit. The 15 percent APRM scram remains active until 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 eight chambers, four in each of the reactor protection system logic channels. The IRM is a five-decade instrument which covers the range of power level between that covered by the SRM and the APRM. The five decades are covered by the IRM by means of a range switch and the five decades are broken down into 10 ranges, each being one-half of a decade in size. The IRM scram setting of 120 divisions is active in each range of the IRM. For example, if the instrument was on range 1, the scram setting would be 120 divisions for that range; likewise if the instrument was on range 5, the scram setting would be 120 divisions for that range.

is greater than the Safety Limit MCPR

2.1 <u>BASES</u> (Cont'd)

<u>IRM Flux Scram Trip Setting (Continued)</u>

Thus, as the IRM is ranged up to accommodate the increase in
power level, the scram setting is also ranged up. A scram at
120 divisions on the IRM instruments remains in effect as long
as the reactor is in the startup mode. In addition, the APRM
15 percent scram prevents higher power operation without being
in the RUN mode. The IRM scram provides protection for changes
which occur both locally and over the entire core. The most
significant sources of reactivity change during the power
increase are due to control rod withdrawal. For insequence
control rod withdrawal, the rate of change of power is slow
enough due to the physical limitation of withdrawing control
rods that heat flux is in equilibrium with the neutron flux. An
IRM scram would result in a reactor shutdown well before any
SAFETY LIMIT is exceeded. For the case of a single control 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. This condition exists at quarter
rod density. Quarter rod density is discussed in
paragraph 7.5.5.4 of the FSAR. 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 1.07. Based
on the above analysis, the IRM provides protection against local
control rod withdrawal errors and continuous withdrawal of
control rods in sequence.

the Safety Limit MCPR

4. <u>Fixed High Neutron Flux Scram Trip</u>

The average power range monitoring (APRM) system, which is
calibrated using heat balance data taken during steady-state
conditions, reads in percent of rated power (3,293 MWt). The
APRM system responds directly to neutron flux. Licensing
analyses have demonstrated that with a neutron flux scram of 120
percent of rated power, none of the abnormal operational
transients analyzed violate the fuel SAFETY LIMIT and there is a
substantial margin from fuel damage.

B. <u>APRM Control Rod Block</u>

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 rod withdrawal beyond a given point at
constant recirculation flow rate and thus prevents scram actuation.
This rod block trip setting, which is automatically varied with
recirculation loop flow rate, prevents an increase in the reactor
power level to excess values due to control rod withdrawal. The
flow variable trip setting is selected to provide adequate margin to
the flow-biased scram setpoint.

2.1 BASES (Cont'd)

C. Reactor Water Low Level Scram and Isolation (Except Main Steam Lines)

The setpoint for the low level scram is above the bottom of the separator skirt. This level has been used in transient analyses dealing with coolant inventory decrease. The results reported in FSAR Subsection 14.5 show that scram and isolation of all process lines (except main steam) at this level adequately protects the fuel and the pressure barrier, because MCPR is greater than 1.07, in all cases, and system pressure does not reach the safety valve settings. The scram setting is sufficiently below normal operating range to avoid spurious scrams.

the Safety Limit MCPR

D. Turbine Stop Valve Closure Scram

The turbine stop valve closure trip anticipates the pressure, neutron flux and heat flux increases that would result from closure of the stop valves. With a trip setting of 10 percent of valve closure from full open, the resultant increase in heat flux is such that adequate thermal margins are maintained even during the worst case transient that assumes the turbine bypass valves remain closed. (Reference 2)

E. Turbine Control Valve Fast Closure or Turbine Trip Scram

Turbine control valve fast closure or turbine trip scram anticipates the pressure, neutron flux, and heat flux increase that could result from control valve fast closure due to load rejection or control valve closure due to turbine trip; each without bypass valve capability. The reactor protection system initiates a scram in less than 30 milliseconds after the start of control valve fast closure due to load rejection or control valve closure due to turbine trip. This scram is achieved by rapidly reducing hydraulic control oil pressure at the main turbine control valve actuator disc dump valves. This loss of pressure is sensed by pressure switches whose contacts form the one-out-of-two-twice logic input to the reactor protection system. This trip setting, a nominally 50 percent greater closure time and a different valve characteristic from that of the turbine stop valve, combine to produce transients very similar to that for the stop valve. No significant change in MCPR occurs. Relevant transient analyses are discussed in References 2 and 3 of the Final Safety Analysis Report. This scram is bypassed when turbine steam flow is below 30 percent of rated, as measured by turbine first state pressure.

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 RBM channels are provided, and one of these may be bypassed from the console for maintenance and/or testing. Automatic rod withdrawal blocks from one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage. 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.

C. Scram Insertion Times

The control rod system is designed to bring the reactor subcritical at a rate fast enough to prevent fuel damage; i.e., to prevent the MCPR from becoming less than 1.07. The limiting power transients are given in Reference 1. Analysis of these transients shows that the negative reactivity rates resulting from the scram with the average response of all drives as given in the above specifications provide the required protection and MCPR remains greater than 1.07.

On an early BWR, some degradation of control rod scram performance occurred during plant STARTUP and was determined to be caused by particulate material (probably construction debris) plugging an internal control rod drive filter. The design of the present control rod drive (Model 7RDB144B) is grossly improved by the relocation of the filter to a location out of the scram drive path; i.e., it can no longer interfere with scram performance, even if completely blocked.

The degraded performance of the original drive (CRD7RDB144A) under dirty operating conditions and the insensitivity of the redesigned drive (CRD7RDB144B) has been demonstrated by a series of engineering tests under simulated reactor operating conditions. The successful performance of the new drive under actual operating conditions has also been demonstrated by consistently good in-service test results for plants using the new drive and may be inferred from plants using the older model

the Safety Limit MCPR

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING

1.1 FUEL CLADDING INTEGRITY

Applicability

Applies to the interrelated variables associated with fuel thermal behavior.

Objective

To establish limits which ensure the integrity of the fuel cladding.

Specification

A. Thermal Power Limits

1. Reactor Pressure >800 psia and Core Flow > 10% of Rated.

When the reactor pressure is greater than 800 psia, the existence of a minimum critical power ratio (MCPR) less than 1.07 1.10 shall constitute violation of the fuel cladding integrity safety limit.

2.1 FUEL CLADDING INTEGRITY

Applicability

Applies to trip settings of the instruments and devices which are provided to prevent the reactor system safety limits from being exceeded.

Objective

To define the level of the process variables at which automatic protective action is initiated to prevent the fuel cladding integrity safety limit from being exceeded.

Specification

The limiting safety system settings shall be as specified below:

A. Neutron Flux Trip Settings

1. APRM Flux Scram Trip Setting (Run Mode) (Flow Biased)

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

1.1 BASES: FUEL CLADDING INTEGRITY SAFETY LIMIT

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

The fuel cladding integrity limit is set such that no calculated fuel damage would occur as a result of an abnormal operational transient. Because fuel damage is not directly observable, the Fuel Cladding Safety Limit is defined with margin to the conditions which would produce onset transition boiling (MCPR of 1.0). ~~This a establishes a Safety Limit such that the minimum critical power ratio (MCPR) is no less than 1.07.~~ MCPR > 1.07 represents a conservative margin relative to the conditions required to maintain fuel cladding integrity.

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. Since boiling transition is not a directly observable parameter, 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, i.e., normal plant operation presented on Figure 2.1-1 by the nominal expected flow control line. The Safety Limit ~~(MCPR of 1.07)~~ has sufficient conservatism to assure that in the event of an abnormal operational transient initiated from a normal operating condition (MCPR > limits specified in Specification 3.5.K) more than 99.9 percent of the fuel rods in the core are expected to avoid boiling transition. The margin between MCPR of 1.0 (onset of transition boiling) and the safety limit, ~~1.07~~ is derived from a detailed statistical analysis considering all of the uncertainties in monitoring the core operating state including uncertainty in the boiling transition correlation as described in Reference 1. The uncertainties employed in deriving the safety limit are provided at the beginning of each fuel cycle.

MCPR

Maintaining the MCPR greater than the Safety Limit MCPR

Because the boiling transition correlation is based on a large
quantity of full scale data there is a very high confidence that
operation of a fuel assembly at the condition of MCPR @ 1.07, would
not produce boiling transition. Thus, although it is not required
to establish the safety limit additional margin exists between the
safety limit and the actual occurrence of loss-of-cladding integrity.

However, if boiling transition were to occur, clad perforation would
not be expected. Cladding temperatures would increase to
approximately 1,100°F which is below the perforation temperature of
the cladding material. This has been verified by tests in the
General Electric Test Reactor (GETR) where fuel similar in design to
BFNP operated above the critical heat flux for a significant period
of time (30 minutes) without clad perforation.

If reactor pressure should ever exceed 1,400 psia during normal
power operation (the limit of applicability of the boiling
transition correlation) it would be assumed that the fuel cladding
integrity Safety Limit has been violated.

At pressures below 800 psia, the core elevation pressure drop
(0 power, 0 flow) is greater than 4.56 psi. At low powers and flows
this pressure differential is maintained in the bypass region of the
core. Since the pressure drop in the bypass region is essentially
all elevation head, the core pressure drop at low power and flows
will always be greater than 4.5 psi. Analyses show that with a flow
of 28×10^3 lbs/hr bundle flow, bundle pressure drop is nearly
independent of bundle power and has a value of 3.5 psi. Thus, the
bundle flow with a 4.56 psi driving head will be greater than
28×10^3 lbs/hr. Full scale ATLAS test data taken at pressures from
14.7 psia to 800 psia indicate that the fuel assembly critical power
at this flow is approximately 3.35 MWt. With the design peaking
factors this corresponds to a core thermal power of more than 50
percent. Thus, a core thermal power limit of 25 percent for reactor
pressures below 800 psia is conservative.

For the fuel in the core 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 water level should drop below
the top of the fuel during this time, the ability to remove decay
heat is reduced. This reduction in cooling capability could lead to
elevated cladding temperatures and clad perforation. As long as the
fuel remains covered with water, sufficient cooling is available to
prevent fuel clad perforation.

equal to the Safety Limit MCPR

Analyses of the limiting transients show that no scram adjustment is required to assure MCPR ~1.07, when the transient is initiated from MCPR limits specified in Specification 3.5.k.

2. **APRM Flux Scram Trip Setting (REFUEL or STARTUP/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 setpoint and the safety limit, 25 percent of rated. The margin is adequate to accommodate anticipated maneuvers associated with power plant startup. Effects of increasing pressure at zero or low void content are minor, cold water from sources available during startup is not much colder than that already in the system, temperature coefficients are small, and control rod patterns are constrained to be uniform by operating procedures backed up by the rod worth minimizer. Worth of individual rods is very low in a uniform rod pattern. Thus, of all possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant 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 approach 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 to assure a scram before the power could exceed the safety limit. The 15 percent APRM scram remains active until 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 eight chambers, four in each of the reactor protection system logic channels. The IRM is a five-decade instrument which covers the range of power level between that covered by the SRM and the APRM. The five decades are covered by the IRM by means of a range switch and the five decades are broken down into 10 ranges, each being one-half of a decade in size. The IRM scram setting of 120 divisions is active in each range of the IRM. For example, if the instrument was on range 1, the scram setting would be 120 divisions for that range; likewise if the instrument was on range 5, the scram setting would be 120 divisions for that range.

is greater than the Safety Limit MCPR

IRM Flux Scram Trip Setting (Continued)

Thus, as the IRM is ranged up to accommodate the increase in power level, the scram setting is also ranged up. A scram at 120 divisions on the IRM instruments remains in effect as long as the reactor is in the startup mode. In addition, the APRM 15 percent scram prevents higher power operation without being in the RUN mode. The IRM scram provides protection for changes which occur both locally and over the entire core. The most significant sources of reactivity change during the power increase are due to control rod withdrawal. For insequence control rod withdrawal, the rate of change of power is slow enough due to the physical limitation of withdrawing control rods that heat flux is in equilibrium with the neutron flux. An IRM scram would result in a reactor shutdown well before any SAFETY LIMIT is exceeded. For the case of a single control 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. This condition exists at quarter rod density. Quarter rod density is discussed in paragraph 7.5.5.4 of the FSAR. 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 1.07. Based on the above analysis, the IRM provides protection against local control rod withdrawal errors and continuous withdrawal of control rods in sequence.

the Safety Limit MCPR

4. Fixed High Neutron Flux Scram Trip

The average power range monitoring (APRM) system, which is calibrated using heat balance data taken during steady-state conditions, reads in percent of rated power (3,293 MWt). The APRM system responds directly to neutron flux. Licensing analyses have demonstrated that with a neutron flux scram of 120 percent of rated power, none of the abnormal operational transients analyzed violate the fuel SAFETY LIMIT and there is a substantial margin from fuel damage.

B. APRM Control Rod Block

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 rod withdrawal beyond a given point at constant recirculation flow rate and thus prevents scram actuation. This rod block trip setting, which is automatically varied with recirculation loop flow rate, prevents an increase in the reactor

2.1 BASES (Cont'd)

power level to excess values due to control rod withdrawal. The flow variable trip setting is selected to provide adequate margin to the flow-biased scram setpoint.

C. **Reactor Water Low Level Scram and Isolation (Except Main Steam Lines)**

The setpoint for the low level scram is above the bottom of the separator skirt. This level has been used in transient analyses dealing with coolant inventory decrease. The results reported in FSAR subsection 14.5 show that scram and isolation of all process lines (except main steam) at this level adequately protects the fuel and the pressure barrier, because MCPR is greater than 1.07, in all cases, and system pressure does not reach the safety valve settings. The scram setting is sufficiently below normal operating range to avoid spurious scrams.

the Safety Limit MCPR

D. **Turbine Stop Valve Closure Scram**

The turbine stop valve closure trip anticipates the pressure, neutron flux and heat flux increases that would result from closure of the stop valves. With a trip setting of 10 percent of valve closure from full open, the resultant increase in heat flux is such that adequate thermal margins are maintained even during the worst case transient that assumes the turbine bypass valves remain closed. (Reference 2)

E. **Turbine Control Valve Fast Closure or Turbine Trip Scram**

Turbine control valve fast closure or turbine trip scram anticipates the pressure, neutron flux, and heat flux increase that could result from control valve fast closure due to load rejection or control valve closure due to turbine trip; each without bypass valve capability. The reactor protection system initiates a scram in less than 30 milliseconds after the start of control valve fast closure due to load rejection or control valve closure due to turbine trip. This scram is achieved by rapidly reducing hydraulic control oil pressure at the main turbine control valve actuator disc dump valves. This loss of pressure is sensed by pressure switches whose contacts form the one-out-of-two-twice logic input to the reactor protection system. This trip setting, a nominally 50 percent greater closure time and a different valve characteristic from that of the turbine stop valve, combine to produce transients very similar to that for the stop valve. No significant change in MCPR occurs. Relevant transient analyses are discussed in References 2 and 3 of the Final Safety Analysis Report. This scram is bypassed when turbine steam flow is below 30 percent of rated, as measured by turbine first state pressure.

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 RBM channels are provided, and one of these may be bypassed from the console for maintenance and/or testing. Automatic rod withdrawal blocks from one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage. 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.

C. Scram Insertion Times

The control rod system is designed to bring the reactor subcritical at a rate fast enough to prevent fuel damage; i.e., to prevent the MCPR from becoming less than 1.07. The limiting power transients are given in Reference 1. Analysis of these transients shows that the negative reactivity rates resulting from the scram with the average response of all drives as given in the above specifications provide the required protection and MCPR remains greater than 1.07.

On an early BWR, some degradation of control rod scram performance occurred during plant STARTUP and was determined to be caused by particulate material (probably construction debris) plugging an internal control rod drive filter. The design of the present control rod drive (Model 7RDB144B) is grossly improved by the relocation of the filter to a location out of the scram drive path; i.e., it can no longer interfere with scram performance, even if completely blocked.

The degraded performance of the original drive (CRD7RDB144A) under dirty operating conditions and the insensitivity of the redesigned drive (CRD7RDB144B) has been demonstrated by a series of engineering tests under simulated reactor operating conditions. The successful performance of the new drive under actual operating conditions has also been demonstrated by consistently good in-service test results for plants using the new drive and may be inferred from plants using the older model

the Safety Limit MCPR

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNITS 1, 2, AND 3

PROPOSED TECHNICAL SPECIFICATION (TS) CHANGE TS-377
REVISED PAGES

I. AFFECTED PAGE LIST

Unit 1	Unit 2	Unit 3
1.1/2.1-1	1.1/2.1-1	1.1/2.1-1
1.1/2.1-8	1.1/2.1-8	1.1/2.1-8
1.1/2.1-9	1.1/2.1-9	1.1/2.1-9
1.1/2.1-13	1.1/2.1-13	1.1/2.1-13
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1.1/2.1-15	1.1/2.1-15	1.1/2.1-15
3.3/4.3-17	3.3/4.3-17	3.3/4.3-17

II. REVISED PAGES

See attached.

* - Denotes spillover page

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING
1.1 FUEL CLADDING INTEGRITY	2.1 FUEL CLADDING INTEGRITY

SAFETY LIMIT

1.1 **FUEL CLADDING INTEGRITY**

Applicability

Applies to the interrelated variables associated with fuel thermal behavior.

Objective

To establish limits which ensure the integrity of the fuel cladding.

Specifications

A. **Thermal Power Limits**

1. Reactor Pressure >800 psia and Core Flow > 10% of Rated.

   When the reactor pressure is greater than 800 psia, the existence of a minimum critical power ratio (MCPR) less than 1.10 shall constitute violation of the fuel cladding integrity safety limit.

LIMITING SAFETY SYSTEM SETTING

2.1 **FUEL CLADDING INTEGRITY**

Applicability

Applies to trip settings of the instruments and devices which are provided to prevent the reactor system safety limits from being exceeded.

Objective

To define the level of the process variables at which automatic protective action is initiated to prevent the fuel cladding integrity safety limit from being exceeded.

Specifications

The limiting safety system settings shall be as specified below:

A. **Neutron Flux Trip Settings**

1. APRM Flux Scram Trip Setting (Run Mode) (Flow biased)

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

1.1 BASES: FUEL CLADDING INTEGRITY SAFETY LIMIT

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

The fuel cladding integrity limit is set such that no calculated fuel damage would occur as a result of an abnormal operational transient. Because fuel damage is not directly observable, the Fuel Cladding Safety Limit is defined with margin to the conditions which would produce onset transition boiling (MCPR of 1.0). Maintaining the MCPR greater than the Safety Limit MCPR represents a conservative margin relative to the conditions required to maintain fuel cladding integrity.

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. Since boiling transition is not a directly observable parameter, 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, i.e., normal plant operation presented on Figure 2.1-1 by the nominal expected flow control line. The Safety Limit has sufficient conservatism to assure that in the event of an abnormal operational transient initiated from a normal operating condition (MCPR > limits specified in Specification 3.5.K) more than 99.9 percent of the fuel rods in the core are expected to avoid boiling transition. The margin between MCPR of 1.0 (onset of transition boiling) and the Safety Limit MCPR is derived from a detailed statistical analysis considering all of the uncertainties in monitoring the core operating state including uncertainty in the boiling transition correlation as described in Reference 1. The uncertainties employed in deriving the safety limit are provided at the beginning of each fuel cycle.

1.1 BASES (Cont'd)

Because the boiling transition correlation is based on a large quantity of full scale data there is a very high confidence that operation of a fuel assembly at the condition of MCPR equal to the Safety Limit MCPR would not produce boiling transition. Thus, although it is not required to establish the safety limit additional margin exists between the safety limit and the actual occurrence of loss of cladding integrity.

However, if boiling transition were to occur, clad perforation would not be expected. Cladding temperatures would increase to approximately 1100°F which is below the perforation temperature of the cladding material. This has been verified by tests in the General Electric Test Reactor (GETR) where fuel similar in design to BFNP operated above the critical heat flux for a significant period of time (30 minutes) without clad perforation.

If reactor pressure should ever exceed 1400 psia during normal power operation (the limit of applicability of the boiling transition correlation) it would be assumed that the fuel cladding integrity Safety Limit has been violated.

At pressures below 800 psia, the core elevation pressure drop (0 power, 0 flow) is greater than 4.56 psi. At low powers and flows this pressure differential is maintained in the bypass region of the core. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low powers and flow will always be greater than 4.56 psi. Analyses show that with a flow of 28×10^3 lbs/hr bundle flow, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.56 psi driving head will be greater than 28×10^3 lbs/hr. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors this corresponds to a core thermal power of more than 50 percent. Thus, a core thermal power limit of 25 percent for reactor pressures below 800 psia is conservative.

For the fuel in the core 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 water level should drop below the top of the fuel during this time, the ability to remove decay heat is reduced. This reduction in cooling capability could lead to elevated cladding temperatures and clad perforation. As long as the fuel remains covered with water, sufficient cooling is available to prevent fuel clad perforation.

2.1 **BASES** (Cont'd)

Analyses of the limiting transients show that no scram adjustment is required to assure MCPR is greater than the Safety Limit MCPR when the transient is initiated from MCPR limits specified in Specification 3.5.k.

2.   APRM Flux Scram Trip Setting (REFUEL or STARTUP/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 setpoint and the safety limit, 25 percent of rated.   The margin is adequate to accommodate anticipated maneuvers associated with power plant startup.   Effects of increasing pressure at zero or low void content are minor, cold water from sources available during startup is not much colder than that already in the system, temperature coefficients are small, and control rod patterns are constrained to be uniform by operating procedures backed up by the rod worth minimizer.   Thus, of all possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant 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 approach 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 to assure a scram before the power could exceed the safety limit.   The 15 percent APRM scram remains active until 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 covered by the IRM by means of a range switch and the 5 decades are broken down into 10 ranges, each being one-half of a decade in size.   The IRM scram 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 at 120 divisions for that range; likewise if the instrument was on range 5, the scram setting would be 120 divisions on that range.

2.1 BASES (Cont'd)

IRM Flux Scram Trip Setting (Continued)

Thus, as the IRM is ranged up to accommodate the increase in power
level, the scram setting is also ranged up. A scram at 120 divisions
on the IRM instruments remains in effect as long as the reactor is in
the startup mode. In addition, the APRM 15 percent scram prevents
higher power operation without being in the RUN mode. The IRM scram
provides protection for changes which occur both locally and over the
entire core. The most significant sources of reactivity change during
the power increase are due to control rod withdrawal. For insequence
control rod withdrawal, the rate of change of power is slow enough due
to the physical limitation of withdrawing control rods that heat flux
is in equilibrium with the neutron flux. An IRM scram would result in
a reactor shutdown well before any SAFETY LIMIT is exceeded. For the
case of a single control 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. This condition exists at quarter rod
density. Quarter rod density is illustrated in paragraph 7.5.5 of the
FSAR. 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 Safety Limit MCPR. Based on the above analysis, the IRM
provides protection against local control rod withdrawal errors and
continuous withdrawal of control rods in sequence.

4. Fixed High Neutron Flux Scram Trip

The average power range monitoring (APRM) system, which is calibrated
using heat balance data taken during steady-state conditions, reads in
percent of rated power (3,293 MWt). The APRM system responds directly
to neutron flux. Licensing analyses have demonstrated that with a
neutron flux scram of 120 percent of rated power, none of the abnormal
operational transients analyzed violate the fuel SAFETY LIMIT and there
is a substantial margin from fuel damage.

B. APRM Control Rod Block

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 rod withdrawal beyond a given point at constant recirculation flow
rate and thus prevents scram actuation. This rod block trip setting, which
is automatically varied with recirculation loop flow rate, prevents an
increase in the reactor power level to excess values due to control rod
withdrawal. The flow variable trip setting is selected to provide adequate
margin to the flow-biased scram setpoint.

2.1 BASES (Cont'd)

C. Reactor Water Low Level Scram and Isolation (Except Main Steam Lines)

The setpoint for the low level scram is above the bottom of the separator skirt. This level has been used in transient analyses dealing with coolant inventory decrease. The results reported in FSAR subsection 14.5 show that scram and isolation of all process lines (except main steam) at this level adequately protects the fuel and the pressure barrier, because MCPR is greater than the Safety Limit MCPR in all cases, and system pressure does not reach the safety valve settings. The scram setting is sufficiently below normal operating range to avoid spurious scrams.

D. Turbine Stop Valve Closure Scram

The turbine stop valve closure trip anticipates the pressure, neutron flux and heat flux increases that would result from closure of the stop valves. With a trip setting of 10 percent of valve closure from full open, the resultant increase in heat flux is such that adequate thermal margins are maintained even during the worst case transient that assumes the turbine bypass valves remain closed. (Reference 2)

E. Turbine Control Valve Fast Closure or Turbine Trip Scram

Turbine control valve fast closure or turbine trip scram anticipates the pressure, neutron flux, and heat flux increase that could result from control valve fast closure due to load rejection or control valve closure due to turbine trip; each without bypass valve capability. The reactor protection system initiates a scram in less than 30 milliseconds after the start of control valve fast closure due to load rejection or control valve closure due to turbine trip. This scram is achieved by rapidly reducing hydraulic control oil pressure at the main turbine control valve actuator disc dump valves. This loss of pressure is sensed by pressure switches whose contacts form the one-out-of-two-twice logic input to the reactor protection system. This trip setting, a nominally 50 percent greater closure time and a different valve characteristic from that of the turbine stop valve, combine to produce transients very similar to that for the stop valve. No significant change in MCPR occurs. Relevant transient analyses are discussed in References 2 and 3 of the Final Safety Analysis Report. This scram is bypassed when turbine steam flow is below 30 percent of rated, as measured by turbine first state pressure.

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 RBM channels are provided, and one of these may be bypassed from the console for maintenance and/or testing. Automatic rod withdrawal blocks from one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage. 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.

C. Scram Insertion Times

The control rod system is designated to bring the reactor subcritical at the rate fast enough to prevent fuel damage; i.e., to prevent the MCPR from becoming less than the Safety Limit MCPR. The limiting power transient is given in Reference 1. Analysis of this transient shows that the negative reactivity rates resulting from the scram with the average response of all the drives as given in the above specification provide the required protection, and MCPR remains greater than the Safety Limit MCPR.

On an early BWR, some degradation of control rod scram performance occurred during plant startup and was determined to be caused by particulate material (probably construction debris) plugging an internal control rod drive filter. The design of the present control rod drive (Model 7RDB144B) is grossly improved by the relocation of the filter to a location out of the scram drive path; i.e., it can no longer interfere with scram performance, even if completely blocked.

The degraded performance of the original drive (CRD7RDB144A) under dirty operating conditions and the insensitivity of the redesigned drive (CRD7RDB144B) has been demonstrated by a series of engineering tests under simulated reactor operating conditions. The successful performance of the new drive under actual operating conditions has also been demonstrated by consistently good in-service test results for plants using the new drive and may be inferred from plants using the older model

1.1/2.1 FUEL CLADDING INTEGRITY

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING

1.1 FUEL CLADDING INTEGRITY

Applicability

Applies to the interrelated variables associated with fuel thermal behavior.

Objective

To establish limits which ensure the integrity of the fuel cladding.

Specifications

A. Thermal Power Limits

1. Reactor Pressure >800 psia and Core Flow > 10% of Rated.

When the reactor pressure is greater than 800 psia, the existence of a minimum critical power ratio (MCPR) less than 1.10 shall constitute violation of the fuel cladding integrity safety limit.

2.1 FUEL CLADDING INTEGRITY

Applicability

Applies to trip settings of the instruments and devices which are provided to prevent the reactor system safety limits from being exceeded.

Objective

To define the level of the process variables at which automatic protective action is initiated to prevent the fuel cladding integrity safety limit from being exceeded.

Specifications

The limiting safety system settings shall be as specified below:

A. Neutron Flux Trip Settings

1. APRM Flux Scram Trip Setting (RUN Mode) (Flow Biased)

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

1.1 BASES: FUEL CLADDING INTEGRITY SAFETY LIMIT

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

The fuel cladding integrity limit is set such that no calculated fuel damage would occur as a result of an abnormal operational transient. Because fuel damage is not directly observable, the Fuel Cladding Safety Limit is defined with margin to the conditions which would produce onset transition boiling (MCPR of 1.0). Maintaining the MCPR greater than the Safety Limit MCPR represents a conservative margin relative to the conditions required to maintain fuel cladding integrity.

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. Since boiling transition is not a directly observable parameter, 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, i.e., normal plant operation presented on Figure 2.1-1 by the nominal expected flow control line. The Safety Limit has sufficient conservatism to assure that in the event of an abnormal operational transient initiated from a normal operating condition (MCPR > limits specified in Specification 3.5.K) more than 99.9 percent of the fuel rods in the core are expected to avoid boiling transition. The margin between MCPR of 1.0 (onset of transition boiling) and the Safety Limit MCPR is derived from a detailed statistical analysis considering all of the uncertainties in monitoring the core operating state including uncertainty in the boiling transition correlation as described in Reference 1. The uncertainties employed in deriving the safety limit are provided at the beginning of each fuel cycle.

1.1 BASES (Cont'd)

Because the boiling transition correlation is based on a large quantity of full scale data there is a very high confidence that operation of a fuel assembly at the condition of MCPR equal to the Safety Limit MCPR would not produce boiling transition. Thus, although it is not required to establish the safety limit additional margin exists between the safety limit and the actual occurrence of loss of cladding integrity.

However, if boiling transition were to occur, clad perforation would not be expected. Cladding temperatures would increase to approximately 1,100°F which is below the perforation temperature of the cladding material. This has been verified by tests in the General Electric Test Reactor (GETR) where fuel similar in design to BFNP operated above the critical heat flux for a significant period of time (30 minutes) without clad perforation.

If reactor pressure should ever exceed 1,400 psia during normal power operation (the limit of applicability of the boiling transition correlation) it would be assumed that the fuel cladding integrity Safety Limit has been violated.

At pressures below 800 psia, the core elevation pressure drop (0 power, 0 flow) is greater than 4.56 psi. At low powers and flows this pressure differential is maintained in the bypass region of the core. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be greater than 4.5 psi. Analyses show that with a flow of 28×10^3 lbs/hr bundle flow, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.56 psi driving head will be greater than 28×10^3 lbs/hr. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors this corresponds to a core thermal power of more than 50 percent. Thus, a core thermal power limit of 25 percent for reactor pressures below 800 psia is conservative.

For the fuel in the core 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 water level should drop below the top of the fuel during this time, the ability to remove decay heat is reduced. This reduction in cooling capability could lead to elevated cladding temperatures and clad perforation. As long as the fuel remains covered with water, sufficient cooling is available to prevent fuel clad perforation.

2.1 BASES (Cont'd)

Analyses of the limiting transients show that no scram adjustment is required to assure MCPR is greater than the Safety Limit MCPR when the transient is initiated from MCPR limits specified in Specification 3.5.k.

2. APRM Flux Scram Trip Setting (REFUEL or STARTUP/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 setpoint and the safety limit, 25 percent of rated. The margin is adequate to accommodate anticipated maneuvers associated with power plant startup. Effects of increasing pressure at zero or low void content are minor, cold water from sources available during startup is not much colder than that already in the system, temperature coefficients are small, and control rod patterns are constrained to be uniform by operating procedures backed up by the rod worth minimizer. Worth of individual rods is very low in a uniform rod pattern. Thus, of all possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant 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 approach to the scram level, the rate of power rise is no more than five percent of rated power per minute, and the APRM system would be more than adequate to assure a scram before the power could exceed the safety limit. The 15 percent APRM scram remains active until 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 eight chambers, four in each of the reactor protection system logic channels. The IRM is a five-decade instrument which covers the range of power level between that covered by the SRM and the APRM. The five decades are covered by the IRM by means of a range switch and the five decades are broken down into 10 ranges, each being one-half of a decade in size. The IRM scram setting of 120 divisions is active in each range of the IRM. For example, if the instrument was on range 1, the scram setting would be 120 divisions for that range; likewise if the instrument was on range 5, the scram setting would be 120 divisions for that range.

2.1 BASES (Cont'd)

IRM Flux Scram Trip Setting (Continued)

Thus, as the IRM is ranged up to accommodate the increase in power
level, the scram setting is also ranged up. A scram at 120 divisions
on the IRM instruments remains in effect as long as the reactor is in
the startup mode. In addition, the APRM 15 percent scram prevents
higher power operation without being in the RUN mode. The IRM scram
provides protection for changes which occur both locally and over the
entire core. The most significant sources of reactivity change during
the power increase are due to control rod withdrawal. For insequence
control rod withdrawal, the rate of change of power is slow enough due
to the physical limitation of withdrawing control rods that heat flux
is in equilibrium with the neutron flux. An IRM scram would result in
a reactor shutdown well before any SAFETY LIMIT is exceeded. For the
case of a single control 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. This condition exists at quarter rod
density. Quarter rod density is discussed in paragraph 7.5.5.4 of the
FSAR. 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 Safety Limit MCPR. Based on the above analysis, the IRM
provides protection against local control rod withdrawal errors and
continuous withdrawal of control rods in sequence.

4. Fixed High Neutron Flux Scram Trip

The average power range monitoring (APRM) system, which is calibrated
using heat balance data taken during steady-state conditions, reads in
percent of rated power (3,293 MWt). The APRM system responds directly
to neutron flux. Licensing analyses have demonstrated that with a
neutron flux scram of 120 percent of rated power, none of the abnormal
operational transients analyzed violate the fuel SAFETY LIMIT and there
is a substantial margin from fuel damage.

B. APRM Control Rod Block

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 rod withdrawal beyond a given point at constant recirculation flow
rate and thus prevents scram actuation. This rod block trip setting, which
is automatically varied with recirculation loop flow rate, prevents an
increase in the reactor power level to excess values due to control rod
withdrawal. The flow variable trip setting is selected to provide adequate
margin to the flow-biased scram setpoint.

2.1 BASES (Cont'd)

C. Reactor Water Low Level Scram and Isolation (Except Main Steam Lines)

The setpoint for the low level scram is above the bottom of the separator skirt. This level has been used in transient analyses dealing with coolant inventory decrease. The results reported in FSAR Subsection 14.5 show that scram and isolation of all process lines (except main steam) at this level adequately protects the fuel and the pressure barrier, because MCPR is greater than the Safety Limit MCPR in all cases, and system pressure does not reach the safety valve settings. The scram setting is sufficiently below normal operating range to avoid spurious scrams.

D. Turbine Stop Valve Closure Scram

The turbine stop valve closure trip anticipates the pressure, neutron flux and heat flux increases that would result from closure of the stop valves. With a trip setting of 10 percent of valve closure from full open, the resultant increase in heat flux is such that adequate thermal margins are maintained even during the worst case transient that assumes the turbine bypass valves remain closed. (Reference 2)

E. Turbine Control Valve Fast Closure or Turbine Trip Scram

Turbine control valve fast closure or turbine trip scram anticipates the pressure, neutron flux, and heat flux increase that could result from control valve fast closure due to load rejection or control valve closure due to turbine trip; each without bypass valve capability. The reactor protection system initiates a scram in less than 30 milliseconds after the start of control valve fast closure due to load rejection or control valve closure due to turbine trip. This scram is achieved by rapidly reducing hydraulic control oil pressure at the main turbine control valve actuator disc dump valves. This loss of pressure is sensed by pressure switches whose contacts form the one-out-of-two-twice logic input to the reactor protection system. This trip setting, a nominally 50 percent greater closure time and a different valve characteristic from that of the turbine stop valve, combine to produce transients very similar to that for the stop valve. No significant change in MCPR occurs. Relevant transient analyses are discussed in References 2 and 3 of the Final Safety Analysis Report. This scram is bypassed when turbine steam flow is below 30 percent of rated, as measured by turbine first state pressure.

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 RBM channels are provided, and one of these may be bypassed from the console for maintenance and/or testing. Automatic rod withdrawal blocks from one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage. 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.

C.   Scram Insertion Times

The control rod system is designed to bring the reactor subcritical at a rate fast enough to prevent fuel damage; i.e., to prevent the MCPR from becoming less than the Safety Limit MCPR. The limiting power transients are given in Reference 1. Analysis of these transients shows that the negative reactivity rates resulting from the scram with the average response of all drives as given in the above specifications provide the required protection and MCPR remains greater than the Safety Limit MCPR.

On an early BWR, some degradation of control rod scram performance occurred during plant STARTUP and was determined to be caused by particulate material (probably construction debris) plugging an internal control rod drive filter. The design of the present control rod drive (Model 7RDB144B) is grossly improved by the relocation of the filter to a location out of the scram drive path; i.e., it can no longer interfere with scram performance, even if completely blocked.

The degraded performance of the original drive (CRD7RDB144A) under dirty operating conditions and the insensitivity of the redesigned drive (CRD7RDB144B) has been demonstrated by a series of engineering tests under simulated reactor operating conditions. The successful performance of the new drive under actual operating conditions has also been demonstrated by consistently good in-service test results for plants using the new drive and may be inferred from plants using the older model

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING

1.1 FUEL CLADDING INTEGRITY

Applicability

Applies to the interrelated variables associated with fuel thermal behavior.

Objective

To establish limits which ensure the integrity of the fuel cladding.

Specification

A. Thermal Power Limits

   1. Reactor Pressure >800 psia and Core Flow > 10% of Rated.

     When the reactor pressure is greater than 800 psia, the existence of a minimum critical power ratio (MCPR) less than 1.10 shall constitute violation of the fuel cladding integrity safety limit.

2.1 FUEL CLADDING INTEGRITY

Applicability

Applies to trip settings of the instruments and devices which are provided to prevent the reactor system safety limits from being exceeded.

Objective

To define the level of the process variables at which automatic protective action is initiated to prevent the fuel cladding integrity safety limit from being exceeded.

Specification

The limiting safety system settings shall be as specified below:

A. Neutron Flux Trip Settings

   1. APRM Flux Scram Trip Setting (Run Mode) (Flow Biased)

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

1.1 BASES: FUEL CLADDING INTEGRITY SAFETY LIMIT

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

The fuel cladding integrity limit is set such that no calculated fuel damage would occur as a result of an abnormal operational transient. Because fuel damage is not directly observable, the Fuel Cladding Safety Limit is defined with margin to the conditions which would produce onset transition boiling (MCPR of 1.0). Maintaining the MCPR greater than the Safety Limit MCPR represents a conservative margin relative to the conditions required to maintain fuel cladding integrity.

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. Since boiling transition is not a directly observable parameter, 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, i.e., normal plant operation presented on Figure 2.1-1 by the nominal expected flow control line. The Safety Limit has sufficient conservatism to assure that in the event of an abnormal operational transient initiated from a normal operating condition (MCPR > limits specified in Specification 3.5.K) more than 99.9 percent of the fuel rods in the core are expected to avoid boiling transition. The margin between MCPR of 1.0 (onset of transition boiling) and the Safety Limit MCPR is derived from a detailed statistical analysis considering all of the uncertainties in monitoring the core operating state including uncertainty in the boiling transition correlation as described in Reference 1. The uncertainties employed in deriving the safety limit are provided at the beginning of each fuel cycle.

1.1   BASES (Cont'd)

Because the boiling transition correlation is based on a large quantity of
full scale data there is a very high confidence that operation of a fuel
assembly at the condition of MCPR equal to the Safety Limit MCPR would not
produce boiling transition. Thus, although it is not required to establish
the safety limit additional margin exists between the safety limit and the
actual occurrence of loss-of-cladding integrity.

However, if boiling transition were to occur, clad perforation would not be
expected. Cladding temperatures would increase to approximately 1,100°F
which is below the perforation temperature of the cladding material. This
has been verified by tests in the General Electric Test Reactor (GETR)
where fuel similar in design to BFNP operated above the critical heat flux
for a significant period of time (30 minutes) without clad perforation.

If reactor pressure should ever exceed 1,400 psia during normal power
operation (the limit of applicability of the boiling transition
correlation) it would be assumed that the fuel cladding integrity Safety
Limit has been violated.

At pressures below 800 psia, the core elevation pressure drop (0 power,
0 flow) is greater than 4.56 psi. At low powers and flows this pressure
differential is maintained in the bypass region of the core. Since the
pressure drop in the bypass region is essentially all elevation head, the
core pressure drop at low power and flows will always be greater than 4.5
psi. Analyses show that with a flow of 28×10^3 lbs/hr bundle flow, bundle
pressure drop is nearly independent of bundle power and has a value of
3.5 psi. Thus, the bundle flow with a 4.56 psi driving head will be
greater than 28×10^3 lbs/hr. Full scale ATLAS test data taken at pressures
from 14.7 psia to 800 psia indicate that the fuel assembly critical power
at this flow is approximately 3.35 MWt. With the design peaking factors
this corresponds to a core thermal power of more than 50 percent. Thus, a
core thermal power limit of 25 percent for reactor pressures below 800 psia
is conservative.

For the fuel in the core 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 water level should drop below the top of the fuel
during this time, the ability to remove decay heat is reduced. This
reduction in cooling capability could lead to elevated cladding
temperatures and clad perforation. As long as the fuel remains covered
with water, sufficient cooling is available to prevent fuel clad
perforation.

2.1 BASES (Cont'd)

Analyses of the limiting transients show that no scram adjustment is required to assure MCPR is greater than the Safety Limit MCPR when the transient is initiated from MCPR limits specified in Specification 3.5.k.

2. APRM Flux Scram Trip Setting (REFUEL or STARTUP/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 setpoint and the safety limit, 25 percent of rated. The margin is adequate to accommodate anticipated maneuvers associated with power plant startup. Effects of increasing pressure at zero or low void content are minor, cold water from sources available during startup is not much colder than that already in the system, temperature coefficients are small, and control rod patterns are constrained to be uniform by operating procedures backed up by the rod worth minimizer. Worth of individual rods is very low in a uniform rod pattern. Thus, of all possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant 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 approach 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 to assure a scram before the power could exceed the safety limit. The 15 percent APRM scram remains active until 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 eight chambers, four in each of the reactor protection system logic channels. The IRM is a five-decade instrument which covers the range of power level between that covered by the SRM and the APRM. The five decades are covered by the IRM by means of a range switch and the five decades are broken down into 10 ranges, each being one-half of a decade in size. The IRM scram setting of 120 divisions is active in each range of the IRM. For example, if the instrument was on range 1, the scram setting would be 120 divisions for that range; likewise if the instrument was on range 5, the scram setting would be 120 divisions for that range.

2.1 BASES (Cont'd)

IRM Flux Scram Trip Setting (Continued)

Thus, as the IRM is ranged up to accommodate the increase in power
level, the scram setting is also ranged up. A scram at 120 divisions
on the IRM instruments remains in effect as long as the reactor is in
the startup mode. In addition, the APRM 15 percent scram prevents
higher power operation without being in the RUN mode. The IRM scram
provides protection for changes which occur both locally and over the
entire core. The most significant sources of reactivity change during
the power increase are due to control rod withdrawal. For insequence
control rod withdrawal, the rate of change of power is slow enough due
to the physical limitation of withdrawing control rods that heat flux
is in equilibrium with the neutron flux. An IRM scram would result in
a reactor shutdown well before any SAFETY LIMIT is exceeded. For the
case of a single control 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. This condition exists at quarter rod
density. Quarter rod density is discussed in paragraph 7.5.5.4 of the
FSAR. 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 Safety Limit MCPR. Based on the above analysis, the IRM
provides protection against local control rod withdrawal errors and
continuous withdrawal of control rods in sequence.

4. Fixed High Neutron Flux Scram Trip

The average power range monitoring (APRM) system, which is calibrated
using heat balance data taken during steady-state conditions, reads in
percent of rated power (3,293 MWt). The APRM system responds directly
to neutron flux. Licensing analyses have demonstrated that with a
neutron flux scram of 120 percent of rated power, none of the abnormal
operational transients analyzed violate the fuel SAFETY LIMIT and there
is a substantial margin from fuel damage.

B. APRM Control Rod Block

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 rod withdrawal beyond a given point at constant recirculation flow
rate and thus prevents scram actuation. This rod block trip setting, which
is automatically varied with recirculation loop flow rate, prevents an

2.1 BASES (Cont'd)

increase in the reactor power level to excess values due to control rod
withdrawal. The flow variable trip setting is selected to provide adequate
margin to the flow-biased scram setpoint.

C.   Reactor Water Low Level Scram and Isolation (Except Main Steam Lines)

The setpoint for the low level scram is above the bottom of the separator
skirt. This level has been used in transient analyses dealing with coolant
inventory decrease. The results reported in FSAR subsection 14.5 show that
scram and isolation of all process lines (except main steam) at this level
adequately protects the fuel and the pressure barrier, because MCPR is
greater than the Safety Limit MCPR in all cases, and system pressure does
not reach the safety valve settings. The scram setting is sufficiently
below normal operating range to avoid spurious scrams.

D.   Turbine Stop Valve Closure Scram

The turbine stop valve closure trip anticipates the pressure, neutron flux
and heat flux increases that would result from closure of the stop valves.
With a trip setting of 10 percent of valve closure from full open, the
resultant increase in heat flux is such that adequate thermal margins are
maintained even during the worst case transient that assumes the turbine
bypass valves remain closed. (Reference 2)

E.   Turbine Control Valve Fast Closure or Turbine Trip Scram

Turbine control valve fast closure or turbine trip scram anticipates the
pressure, neutron flux, and heat flux increase that could result from
control valve fast closure due to load rejection or control valve closure
due to turbine trip; each without bypass valve capability. The reactor
protection system initiates a scram in less than 30 milliseconds after the
start of control valve fast closure due to load rejection or control valve
closure due to turbine trip. This scram is achieved by rapidly reducing
hydraulic control oil pressure at the main turbine control valve actuator
disc dump valves. This loss of pressure is sensed by pressure switches
whose contacts form the one-out-of-two-twice logic input to the reactor
protection system. This trip setting, a nominally 50 percent greater
closure time and a different valve characteristic from that of the turbine
stop valve, combine to produce transients very similar to that for the stop
valve. No significant change in MCPR occurs. Relevant transient analyses
are discussed in References 2 and 3 of the Final Safety Analysis Report.
This scram is bypassed when turbine steam flow is below 30 percent of
rated, as measured by turbine first state pressure.

3.3/4.3 BASES (Cont'd)

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 RBM channels are provided, and one of these may be bypassed from the console for maintenance and/or testing. Automatic rod withdrawal blocks from one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage. 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.

C. Scram Insertion Times

The control rod system is designed to bring the reactor subcritical at a rate fast enough to prevent fuel damage; i.e., to prevent the MCPR from becoming less than the Safety Limit MCPR. The limiting power transients are given in Reference 1. Analysis of these transients shows that the negative reactivity rates resulting from the scram with the average response of all drives as given in the above specifications provide the required protection and MCPR remains greater than the Safety Limit MCPR.

On an early BWR, some degradation of control rod scram performance occurred during plant STARTUP and was determined to be caused by particulate material (probably construction debris) plugging an internal control rod drive filter. The design of the present control rod drive (Model 7RDB144B) is grossly improved by the relocation of the filter to a location out of the scram drive path; i.e., it can no longer interfere with scram performance, even if completely blocked.

The degraded performance of the original drive (CRD7RDB144A) under dirty operating conditions and the insensitivity of the redesigned drive (CRD7RDB144B) has been demonstrated by a series of engineering tests under simulated reactor operating conditions. The successful performance of the new drive under actual operating conditions has also been demonstrated by consistently good in-service test results for plants using the new drive and may be inferred from plants using the older model

ENCLOSURE 4

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNITS 1, 2, AND 3

REVISED UNIT 2 CYCLE 9 CORE OPERATING LIMITS REPORT