JUN 6 1973

Docket No. 50-269

Duke Power Company
ATTN: Mr. A. C. Thies
Senior Vice President
Production and Transmission
422 South Church Street
P. O. Box 2178
Charlotte, North Carolina 28201

Change No. 3
License No. DPR-38

Gentlemen:

Change No. 1 to the Technical Specifications, Appendix A, to Operating License No. DPR-38 for the Oconee Unit 1 plant, limited power operations to 75% of full rated power for an interim period until we have completed our evaluation of fuel densification.

We have completed our fuel densification evaluation of Oconee Unit 1 based upon the information you have provided and have concluded that Technical Specification changes will be required to compensate for the effects of fuel densification on the thermal behavior of the fuel under normal operation, transient and accident conditions at 100% of full rated power.

In addition, since this is the first nuclear steam supply system of this design to go into service, we feel that some power margin should be held in reserve until the system has performed at significant power levels for a reasonable time and design objectives have been verified.

We have concluded that operation of Oconee Unit 1 with the appropriate limits of the Technical Specifications does not involve significant hazards consideration and that there is reasonable assurance that the health and safety of the public will not be endangered. Accordingly, pursuant to Section 50.59 of 10 CFR Part 50, Appendix A to Operating License No. DPPR-38, Docket No. 50-269

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In addition, since this is the first nuclear steam supply system of this design to go into service, we feel that some power margin should be held in reserve until the system has performed at significant power levels for a reasonable time and design objectives have been verified.

We have concluded that operation of Oconee Unit 1 with the appropriate limits of the Technical Specification does not involve significant hazards considerations not described or implicit in the Final Safety Analysis Report and that there is reasonable assurance that the health and safety of the public will not be endangered. Accordingly, pursuant to Section 50.59 of 10 CFR Part 50, Appendix A to Operating License No. DPFR-38, Technical Specifications 1.6, 1.8, 2.1, 2.3, 3.1.3, 3.5.2, 3.11, 4.1 and 4.7 are revised as the enclosure to this letter. This change will be known as Change No. 3 to the Technical Specifications, License DPR-38 and is effective on date of issuance.

Sincerely,

Original signed by
R. C. DeYoung

R. C. DeYoung, Assistant Director
for Pressurized Water Reactors
Directorate of Licensing

Enclosure:
Change No. 3 to DPR-38

cc: William L. Porter, Esquire
Duke Power Company
422 South Church Street
Charlotte, North Carolina 28201

DISTRIBUTION
AEC PDR
Local PDR
Docket
RP Rdg
PWR-4 Rdg
OGC
RO (3)
RCDeYoung
DSkovholt
RWKlecker
RVollmer
MJinks (w/4 encls)
IAPeltier
ASchwencer
EIGoulbourne
JHendrie

bcc: J. R. Buchanan, ORNL
T. B. Abernathy, DTIE

OFFICE ▶	PWR-4	OGC	DD:TR/L	L:C/PWR-4	L:AD/PWRs	RO
SURNAME ▶	IAPeltier:kmf	XX Wilchins	JHendrie	ASchwencer	RCDeYoung	JKeppler
DATE ▶	6/1/73	6/c/73	6/5/73	6/0/73	6/6/73	6/5/73

1.5.5 Heat Balance Check

A heat balance check is a comparison of the indicated neutron power and core thermal power.

1.5.6 Heat Balance Calibration

An adjustment of the power range channel amplifiers output to agree with the core thermal power as determined by a heat balance on the secondary side of the steam generator considering all heat loses and additions.

1.6 POWER DISTRIBUTION

1.6.1 Quadrant Power Tilt

Quadrant power tilt is defined by the following equation and is expressed in percent.

$$100 \left(\frac{\text{Power in any core quadrant}}{\text{Average power of all quadrants}} - 1 \right)$$

The power in any quadrant is determined from the power range channel displayed on the console for that quadrant. The average power is determined from an average of the outputs of the power range channels. If one of the power range channels is out of service, the incore detectors will be used. The quadrant power tilt limits as a function of power are stated in Specification 3.5.2.4.

1.6.2 Reactor Power Imbalance

Reactor power imbalance is the power in the top half of the core minus the power in the bottom half of the core expressed as a percentage of rated power. Imbalance is monitored continuously by the RPS using input from the power range channels. Imbalance limits are defined in Specification 2.1 and imbalance setpoints are defined in Specification 2.3.

1.7 CONTAINMENT INTEGRITY

Containment integrity exists when the following conditions are satisfied:

a. The equipment hatch is closed and sealed and both doors of the personnel hatch and emergency hatch are closed and sealed except as in b below.

b. At least one door on each of the personnel hatch and emergency hatch is closed and sealed during refueling or personnel passage through these hatches.

c. All non-automatic containment isolation valves and blind flanges are closed as required.

d. All automatic containment isolation valves are operable or locked closed.

e. The containment leakage determined at the last testing interval satisfies Specification 4.4.1.

1.8 ABNORMAL OCCURRENCE

An abnormal occurence means the occurence of any plant condition that:

a. Results in a protective instrumentation setting less conservative than the Limiting Safety System Setting as established in Technical Specifications, or

b. Exceeds a Limiting Condition for Operation as established in the Technical Specifications, or

c. Causes any significant uncontrolled or unplanned release of radioactive material from the site, or

d. Results in abnormal degradation of one of the several boundaries which are designed to contain the radioactive materials resulting from the fission process, or

e. Results in uncontrolled or unanticipated changes in reactivity greater than 1% $\Delta k/k$ except for trip.

1.9 UNUSUAL EVENTS

An unusual event is:

a. Discovery of any substantial errors in the transient or accident analyses, or in the methods used for such analyses, as described in the Safety Analysis Report or in the bases for the Technical Specifications.

b. Any substantial variance from performance specifications contained in the Technical Specifications or the Safety Analysis Report.

c. Any observed inadequacy in the implementation of administrative or procedural controls during the operation of the facility which could significantly affect the safety of operations.

d. Any occurrence resulting in an Engineered Safety System or Reactor Protective System component malfunction or system or component malfunction which could render a safety system incapable of performing its intended safety function.

e. Any occurrence arising from natural or offsite man-made events that affect or threaten to affect the safe operation of the plant.

Date of Issuance: June 8 , 1973

2 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS, REACTOR CORE

Applicability

Applies to reactor thermal power, reactor power imbalance, reactor coolant system pressure, coolant temperature, and coolant flow during power operation of the plant.

Objective

To maintain the integrity of the fuel cladding.

Specification

The combination of the reactor system pressure and coolant temperature shall not exceed the safety limit as defined by the locus of points established in Figure 2.1-1. If the actual pressure/temperature point is below and to the right of the line, the safety limit is exceeded.

The combination of reactor thermal power and reactor power imbalance (power in the top half of the core minus the power in the bottom half of the core expressed as a percentage of the rated power) shall not exceed the safety limit as defined by the locus of points (solid line) for the specified flow set forth in Figure 2.1-2. If the actual-reactor-thermal-power/reactor-power-imbalance point is above the line for the specified flow, the safety limit is exceeded.

Bases

To maintain the integrity of the fuel cladding and to prevent fission product release, it is necessary to prevent overheating of the cladding under normal operation conditions. This is accomplished by operating within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is large enough so that the clad surface temperature is only slightly greater than the coolant temperature. The upper boundary of the nucleate boiling regime is termed "departure from nucleate boiling" (DNB). At this point there is a sharp reduction of the heat transfer coefficient, which would result in high cladding temperatures and the possibility of cladding failure. Although DNB is not an observable parameter during reactor operation, the observable parameters of neutron power, reactor coolant flow, temperature, and pressure can be related to DNB through the use of the W-3 correlation.(1) The W-3 correlation has been developed to predict DNB and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the actual heat flux, is indicative of the margin to DNB. The minimum value of the DNBR, during steady-state operation, normal operational transients, and anticipated transients is limited to 1.3. A DNBR of 1.3 corresponds to a 94.3% probability at a 99% confidence level that DNB will not occur; this is considered a conservative margin to DNB for all operating conditions. The difference between the actual core outlet pressure and the indicated reactor coolant system pressure has been considered in determining the core protection safety limits. The difference in these two pressures is nominally 45 psi; however, only a 30 psi drop was assumed in reducing the pressure trip set points to correspond to the elevated location where the pressure is actually measured.

The curve presented in Figure 2.1-1 represents the conditions at which a minimum DNBR of 1.3 is predicted for the maximum possible thermal power (112%) when four reactor coolant pumps are operating (reactor coolant flow is 131.3 x 10^6 lbs/hr). This curve is based on the following nuclear power peaking factors(2) with potential fuel densification effects;

$$F_q^N = 2.67; \quad F_{\Delta H}^N = 1.78; \quad F_z^N = 1.50$$

The design peaking combination results in a more conservative DNBR than any other shape that exists during normal operation.

The curves of Figure 2.1-2 are based on the more restrictive of two thermal limits and include the effects of potential fuel densification:

1. The 1.3 DNBR limit produced by a nuclear power peaking factor of $F_q^N = 2.67$ or the combination of the radial peak, axial peak and position of the axial peak that yields no less than a 1.3 DNBR.

2. The combination of radial and axial peak that causes central fuel melting at the hot spot. The limit is 20.1 kw/ft.

Power peaking is not a directly observable quantity and therefore limits have been established on the bases of the reactor power imbalance produced by the power peaking.

The specified flow rates for curves 1, 2, 3, and 4 of Figure 2.1-2 correspond to the expected minimum flow rates with four pumps, three pumps, one pump in each loop and two pumps in one loop, respectively.

The curve of Figure 2.1-1 is the most restrictive of all possible reactor coolant pump--maximum thermal power combinations shown in Figure 2.1-3. The curves of Figure 2.1-3 represent the conditions at which a minimum DNBR of 1.3 is predicted at the maximum possible thermal power for the number of reactor coolant pumps in operation or the local quality at the point of minimum DNBR is equal to 15%,(3) whichever condition is more restrictive.

Using a local quality limit of 15% at the point of minimum DNBR as a basis for curves 2 and 4 of Figure 2.1-3 is a conservative criteron even though the quality of the exit is higher than the quality at the point of minimum DNBR.

The DNBR as calculated by the W-3 correlation continually increases from point of minimum DNBR, so that the exit DNBR is 1.7 or higher, depending on the pressure. Extrapolation of the W-3 correlation beyond its published quality range of +15% is justified on the basis of experimental data.(4)

The maximum thermal power for three pump operation is 87% due to a power level trip produced by the flux-flow ratio (75% flow x 1.08 = 81% power) plus the maximum calibration and instrumentation error. The maximum thermal power for other reactor coolant pump conditions are produced in a similar manner.

Date of Issuance: June ,1973

For each curve of Figure 2.1-3, a pressure-temperature point above and to the left of the curve would result in a DNBR greater than 1.3 or a local quality at the point of minimum DNBR less than 15% for that particular reactor coolant pump situation. The 1.3 DNBR curve for four pump operation is more restrictive than any other reactor coolant pump situation because any pressure/temperature point above and to the left of the four pump curve will be above and to the left of the other curves.

REFERENCES

(1) FSAR, Section 3.2.3.1.1

(2) FSAR, Section 3.2.3.1.1.c

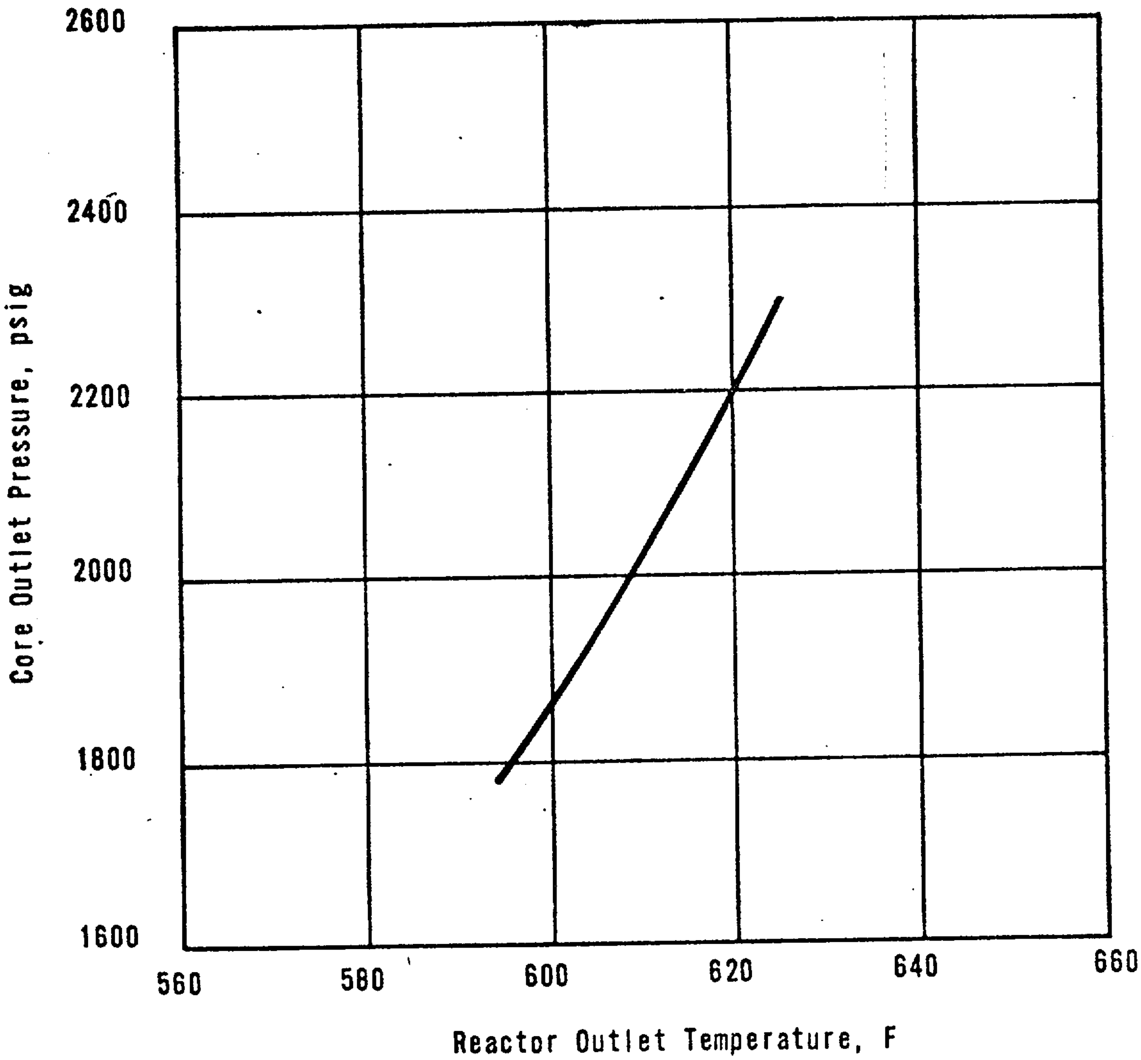
(3) FSAR, Section 3.2.3.1.1.k

(4) The following papers which were presented at the Winter Annual Meeting, ASME, November 18, 1969, during the "Two-phase Flow and Heat Transfer in Rod Bundles Symposium":

(a) Wilson, et.al.
"Critical Heat Flux in Non-Uniform Heater Rod Bundles."

(b) Gellerstedt, et.al.
"Correlation of a Critical Heat Flux in a Bundle Cooled by Pressurized Water."

Date of Issuance: June 6, 1973

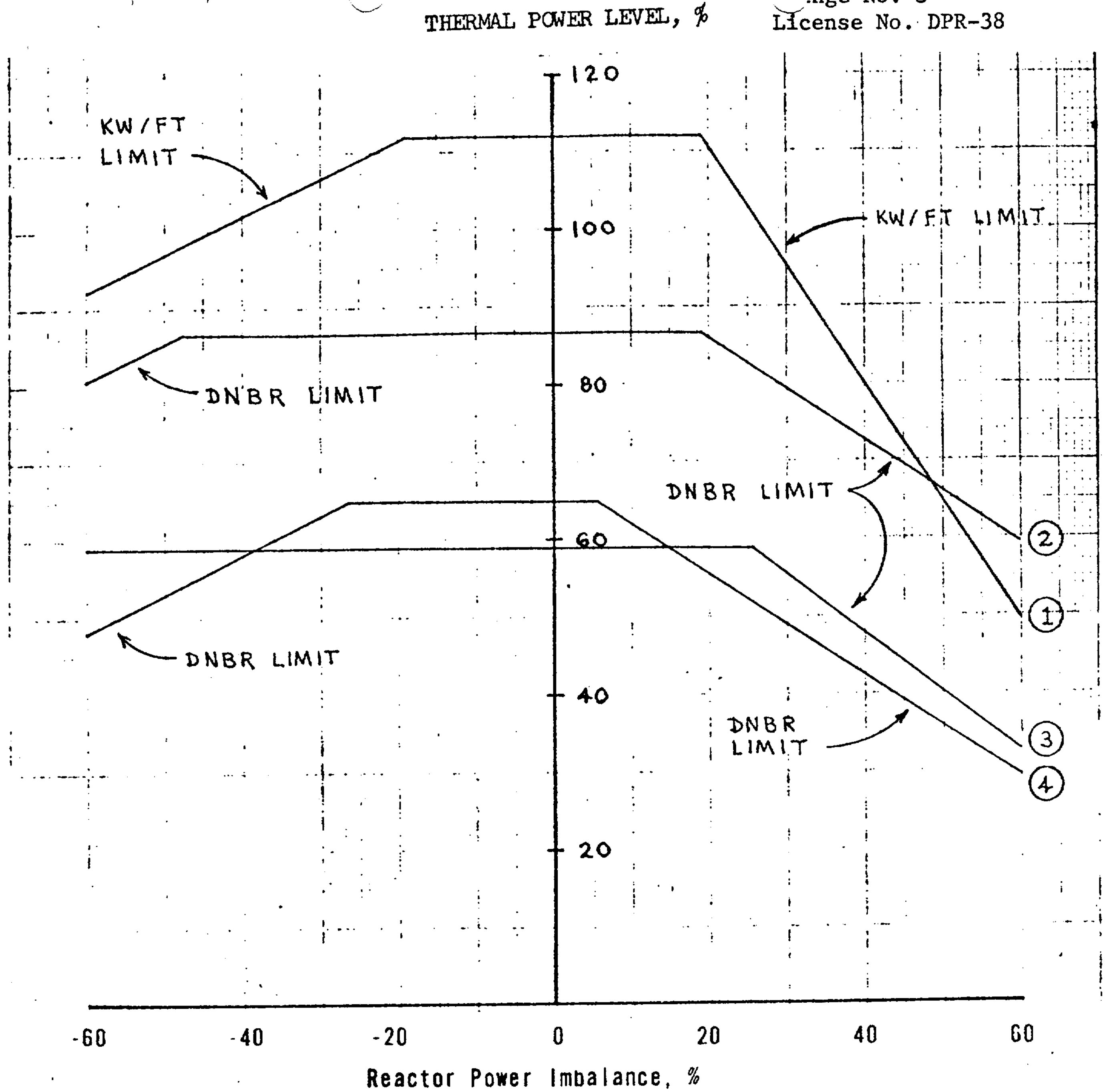


CORE PROTECTION SAFETY LIMIT

Date of Issuance: June 5 , 1973

OCONEE NUCLEAR STATION

Figure 2.1-1



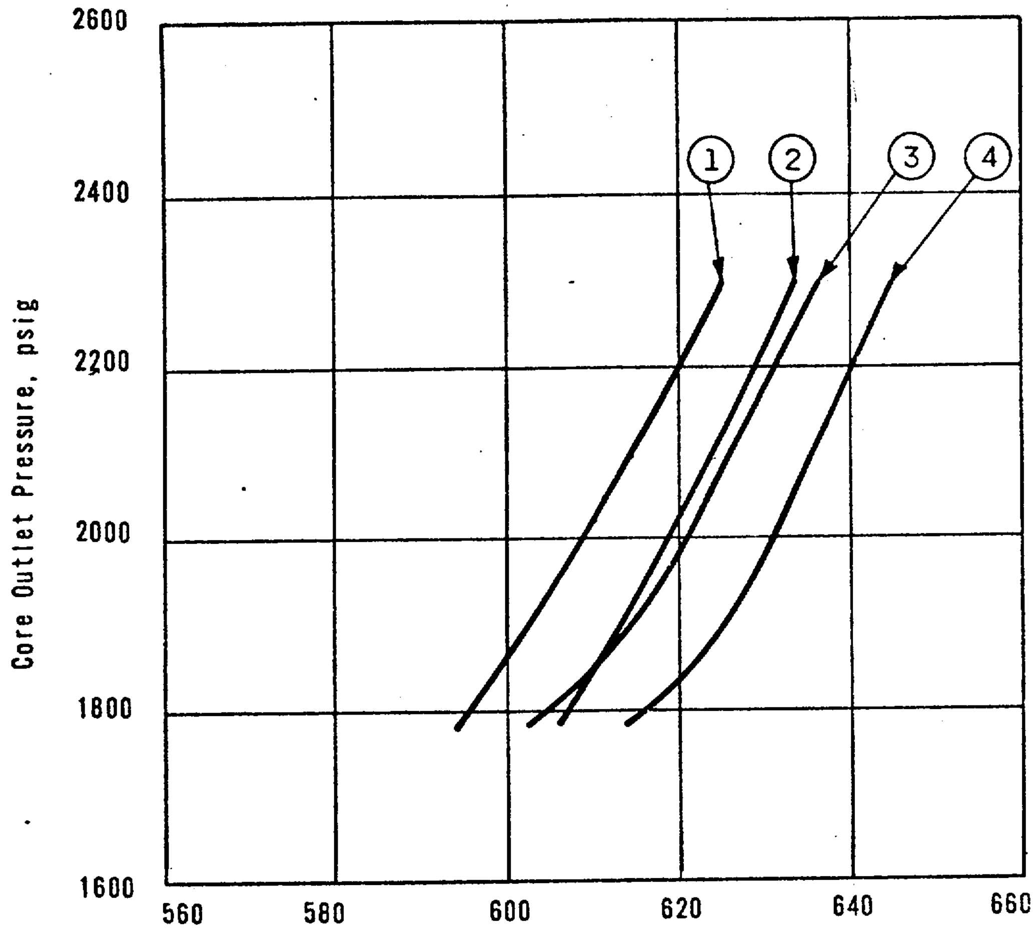
CURVE	REACTOR COOLANT FLOW (LB/HR)
1	131.3×10^6
2	98.1×10^6
3	64.4×10^6
4	60.1×10^6

CORE PROTECTION SAFETY LIMITS

Date of Issuance: June 8, 1973

OCONEE NUCLEAR STATION

Figure 2.1 - 2



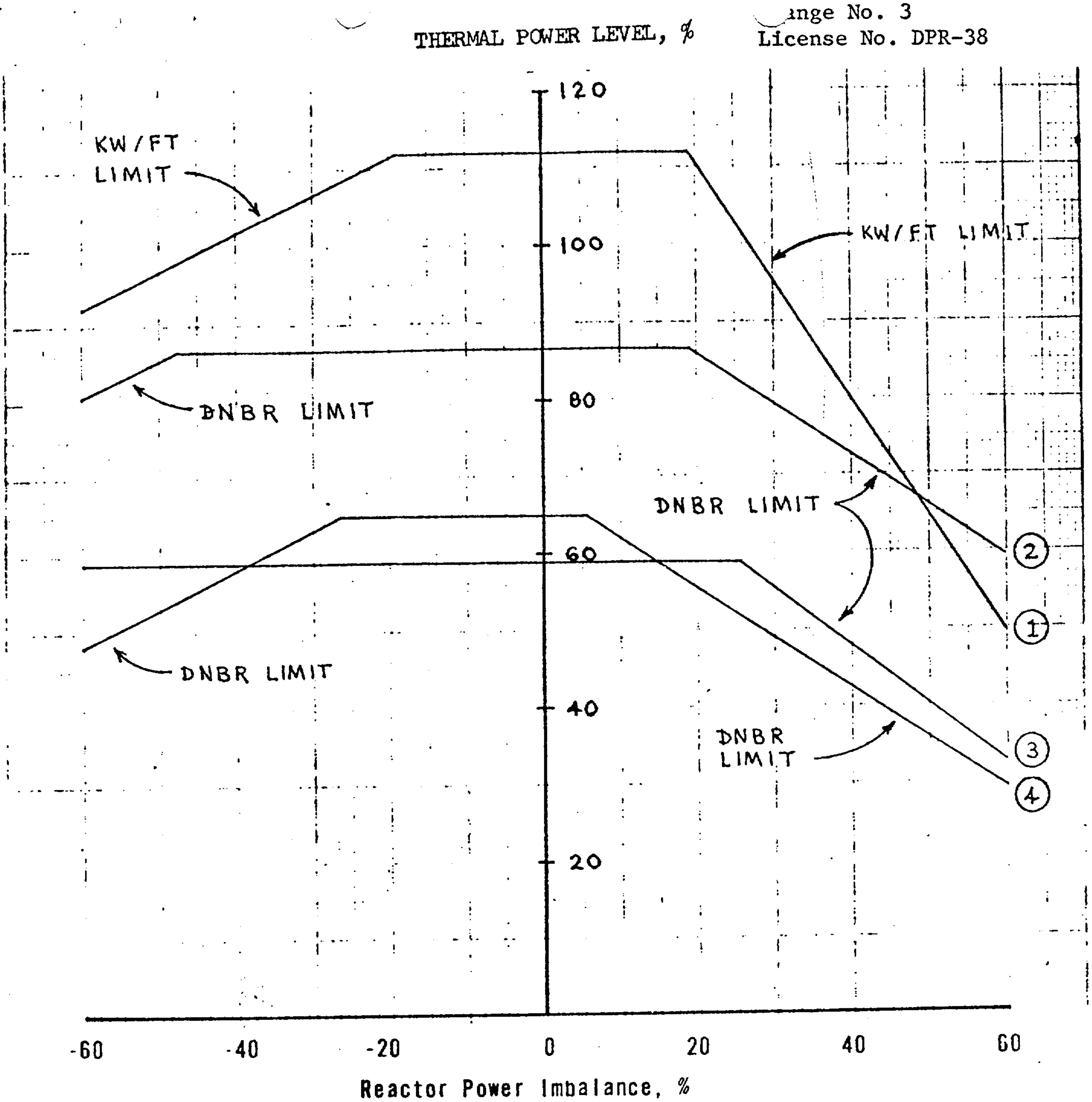
CURVE	REACTOR COOLANT FLOW (LBS/HR)	POWER	PUMPS OPERATING (TYPE OF LIMIT)
1	131.3 x 10^6 (100%)	112%	FOUR PUMPS (DNBR LIMIT)
2	60.1 x 10^6 (45.8%)	65%	TWO PUMPS IN ONE LOOP (QUALITY LIMIT)
3	98.1 x 10^6 (74.7%)	87%	THREE PUMPS (DNBR LIMIT)
4	64.4 x 10^6 (49.0%)	59%	ONE PUMP IN EACH LOOP (QUALITY LIMIT)

CORE PROTECTION SAFETY LIMITS

Date of Issuance: June 8, 1973

OCONEE NUCLEAR STATION

Figure 2.1-3

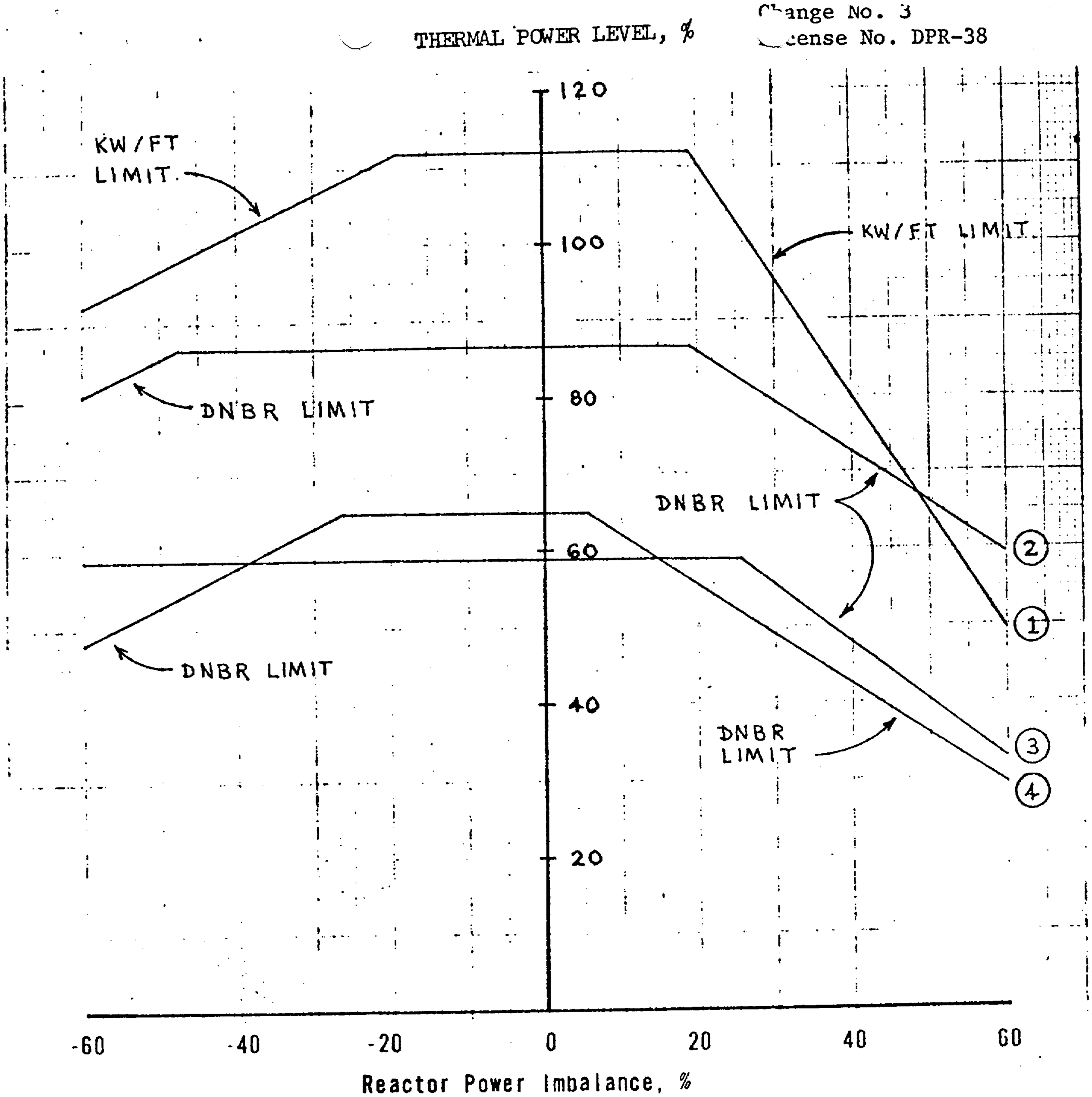


CURVE	REACTOR COOLANT FLOW (LB/HR)
1	131.3×10^6
2	98.1×10^6
3	64.4×10^6
4	60.1×10^6

CORE PROTECTION SAFETY LIMITS

Date of Issuance: June 5, 1973

OCONEE NUCLEAR STATION
Figure 2.1 - 2



CURVE	REACTOR COOLANT FLOW (LB/HR)
1	131.3×10^6
2	98.1×10^6
3	64.4×10^6
4	60.1×10^6

CORE PROTECTION SAFETY LIMITS

OCONEE NUCLEAR STATION

Figure 2.1 - 2

Date of Issuance: June 8, 1973

2.3 LIMITING SAFETY SYSTEM SETTINGS, PROTECTIVE INSTRUMENTATION

Applicability

Applies to instruments monitoring reactor power, reactor power imbalance, reactor coolant system pressure, reactor coolant outlet temperature, flow, number of pumps in operation, and high reactor building pressure.

Objective

To provide automatic protective action to prevent any combination of process variables from exceeding a safety limit.

Specification

The reactor protective system trip setting limits and the permissible by-passes for the instrument channels shall be as stated in Table 2.3-1 and Figure 2.3-2.

The pump monitors shall produce a reactor trip for the following conditions:

a. Loss of two pumps and reactor power level is greater than 55% of rated power.

b. Loss of two pumps in one reactor coolant loop and reactor power level is greater than 0.0% of rated power. (Reactor power level trip setpoint is reset to 55% of rated power for single loop operation.)

c. Loss of one or two pumps during two-pump operation.

Bases

The reactor protective system consists of four instrument channels to monitor each of several selected plant conditions which will cause a reactor trip if any one of these conditions deviates from a pre-selected operating range to the degree that a safety limit may be reached.

The trip setting limits for protective system instrumentation are listed in Table 2.3-1. The safety analysis has been based upon these protective system instrumentation trip set points plus calibration and instrumentation errors.

Nuclear Overpower

A reactor trip at high power level (neutron flux) is provided to prevent damage to the fuel cladding from reactivity excursions too rapid to be detected by pressure and temperature measurements.

During normal plant operation with all reactor coolant pumps operating, reactor trip is initiated when the reactor power level reaches 105.5% of rated power. Adding to this the possible variation in trip set points due to calibration and instrument errors, the maximum actual power at which a trip would be actuated could be 112%, which is more conservative than the value used in the safety analysis.[4]

Date of Issuance: June 8 , 1973

Overpower Trip Based on Flow and Imbalance

The power level trip set point produced by the reactor coolant system flow is based on a power-to-flow ratio which has been established to accommodate the most severe thermal transient considered in the design, the loss-of-coolant flow accident from high power. Analysis has demonstrated that the specified power to flow ratio is adequate to prevent a DNBR of less than 1.3 should a low flow condition exist due to any electrical malfunction.

The power level trip set point produced by the power-to-flow ratio provides both high power level and low flow protection in the event the reactor power level increases or the reactor coolant flow rate decreases. The power level trip set point produced by the power to flow ratio provides overpower DNB protection for all modes of pump operation. For every flow rate there is a maximum permissible power level, and for every power level there is a minimum permissible low flow rate. Typical power level and low flow rate combinations for the pump situations of Table 2.3-1 are as follows:

1. Trip would occur when four reactor coolant pumps are operating if power is 108% and reactor flow rate is 100%, or flow rate is 93% and power level is 100%.

2. Trip would occur when three reactor coolant pumps are operating if power is 81.0% and reactor flow rate is 74.7% or flow rate is 69% and power level is 75%.

3. Trip would occur when two reactor coolant pumps are operating in a single loop if power is 59% and the operating loop flow rate is 54.5% or flow rate is 43% and power level is 46%.

4. Trip would occur when one reactor coolant pump is operating in each loop (total of two pumps operating) if the power is 53% and reactor flow rate is 49.0% or flow rate is 45% and the power level is 49%.

For safety calculations the maximum calibration and instrumentation errors for the power level trip were used.

The power-imbalance boundaries are established in order to prevent reactor thermal limits from being exceeded. These thermal limits are either power peaking kw/ft limits or DNBR limits. The reactor power imbalance (power in the top half of core minus power in the bottom half of core) reduces the power level trip produced by the power-to-flow ratio such that the boundaries of Figure 2.3-2 are produced. The power-to-flow ratio reduces the power level trip and associated reactor power-reactor power-imbalance boundaries by 1.08% for a 1% flow reduction.

Pump Monitors

The pump monitors prevent the minimum core DNBR from decreasing below 1.3 by tripping the reactor due to the loss of reactor coolant pump(s). The circuitry monitoring pump operational status provides redundant trip protection for DNB by tripping the reactor on a signal diverse from that of the power-to-flow ratio. The pump monitors also restrict the power level for the number of pumps in operation.

Reactor Coolant System Pressure

During a startup accident from low power or a slow rod withdrawal from high power, the system high pressure set point is reached before the nuclear overpower trip set point. The trip setting limit shown in Figure 2.3-1 for high reactor coolant system pressure (2355 psig) has been established to maintain the system pressure below the safety limit (2750 psig) for any design transient.(1)

The low pressure (1800 psig) and variable low pressure ($16.25T_{out} - 7769$) trip setpoint shown in Figure 2.3-1 have been established to maintain the DNB ratio greater than or equal to 1.3 for those design accidents that result in a pressure reduction.(2,3)

Due to the calibration and instrumentation errors the safety analysis used a variable low reactor coolant system pressure trip value of ($16.25T_{out} - 7809$).

Coolant Outlet Temperature

The high reactor coolant outlet temperature trip setting limit (619 F) shown in Figure 2.3-1 has been established to prevent excessive core coolant temperatures in the operating range. Due to calibration and instrumentation errors, the safety analysis used a trip set point of 620 F.

Reactor Building Pressure

The high reactor building pressure trip setting limit (4 psig) provides positive assurance that a reactor trip will occur in the unlikely event of a loss-of-coolant accident, even in the absence of a low reactor coolant system pressure trip.

Shutdown Bypass

In order to provide for control rod drive tests, zero power physics testing, and startup procedures, there is provision for bypassing certain segments of the reactor protection system. The reactor protection system segments which can be bypassed are shown in Table 2.3-1. Two conditions are imposed when the bypass is used:

1. By administrative control the nuclear overpower trip set point must be reduced to a value $\leq$ 5.0% of rated power during reactor shutdown.

2. A high reactor coolant system pressure trip set point of 1720 psig is automatically imposed.

The purpose of the 1720 psig high pressure trip set point is to prevent normal operation with part of the reactor protection system bypassed. This high pressure trip set point is lower than the normal low pressure trip set point so that the reactor must be tripped before the bypass is initiated. The over power trip set point of $\leq$ 5.0% prevents any significant reactor power from being produced when performing the physics tests. Sufficient natural circulation(5) would be available to remove 5.0% of rated power if none of the reactor coolant pumps were operating.

Date of Issuance: June 3, 1973

Single Loop Operation

Single loop operation is permitted only after the reactor has been tripped. After the pump contact monitor trip has occurred the following actions will permit single loop operation:

1. Reset the pump contact monitor power level trip set point to 55.0%.

2. Trip one of the two protective channels receiving outlet temperature information from sensors in the idle loop.

Tripping one of the two protection channels receiving outlet temperature information from the idle loop assures a protective system trip logic of one out of two.

REFERENCES

(1) FSAR, Section 14.1.2.2

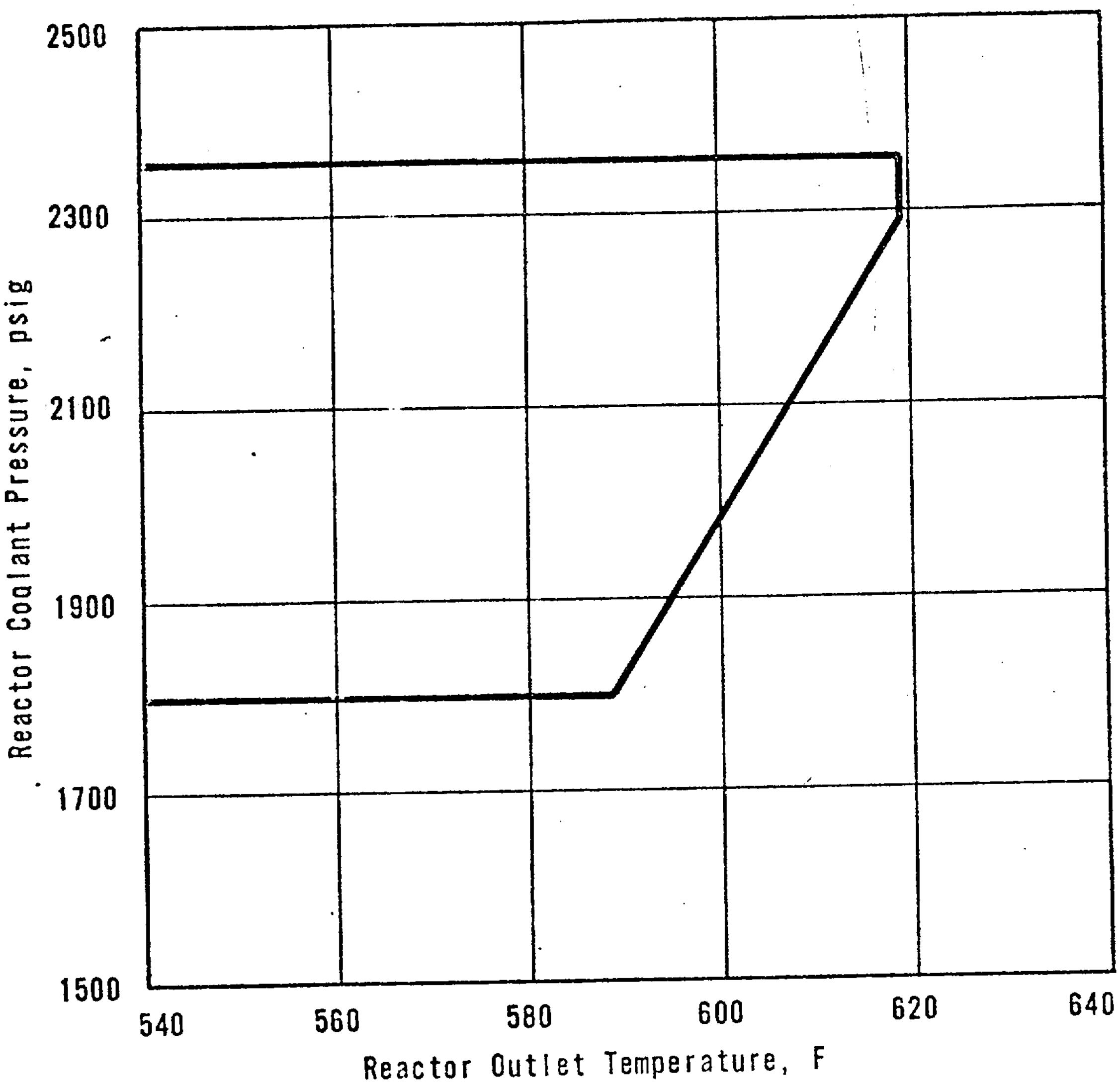
(2) FSAR, Section 14.1.2.7

(3) FSAR, Section 14.1.2.8

(4) FSAR, Section 14.1.2.3

(5) FSAR, Section 14.1.2.6

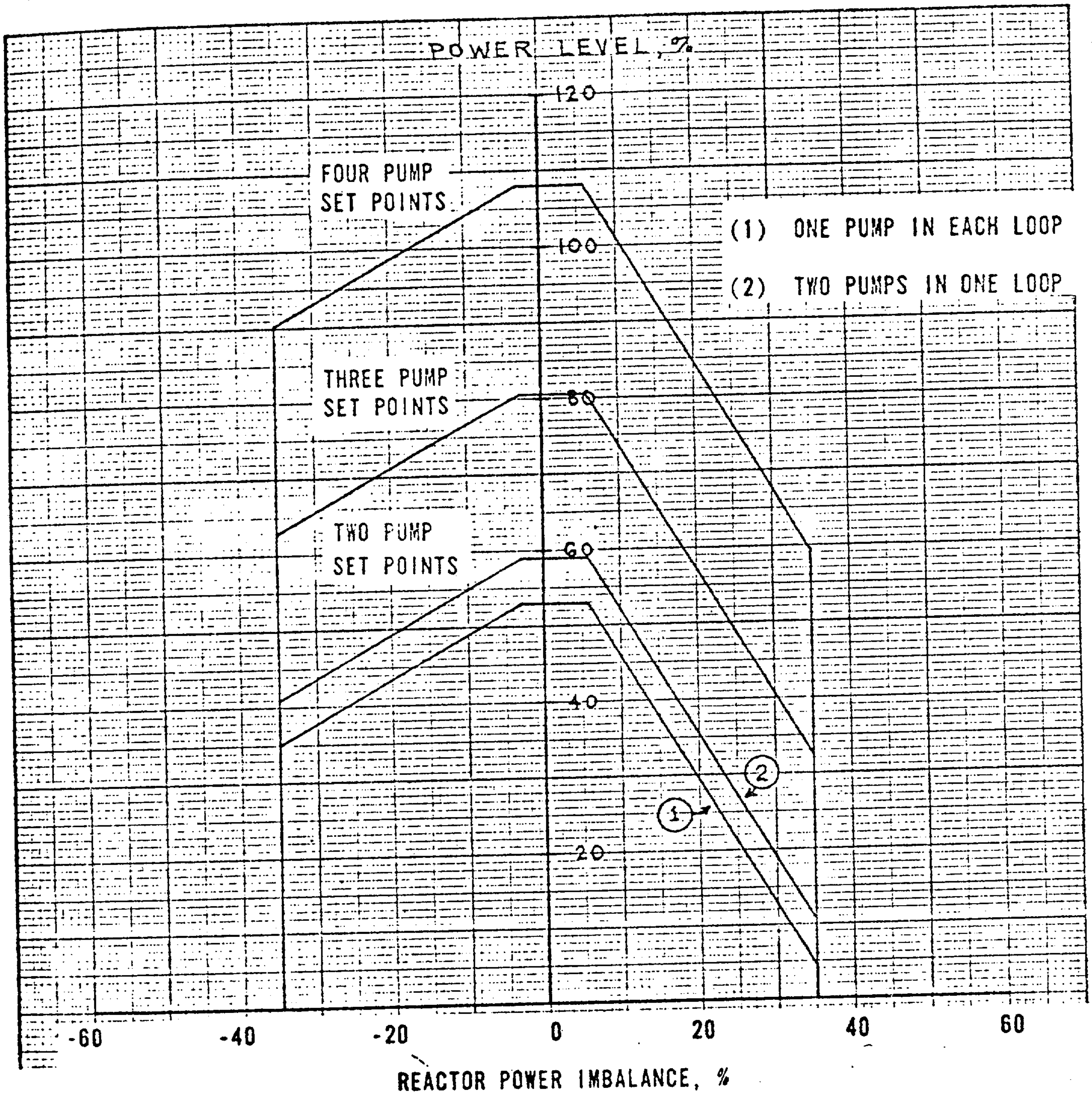
Date of Issuance: June 6 , 1973



PROTECTIVE SYSTEM MAXIMUM
ALLOWABLE SET POINTS

Date of Issuance: June 8, 1973

OCONEE NUCLEAR STATION
Figure 2.3 - 1



PROTECTIVE SYSTEM MAXIMUM ALLOWABLE SET POINTS

OCONEE NUCLEAR STATION
Figure 2.3 - 2

Date of Issuance: June 3 , 1973

Date of Issuance: June , 1973

2.3-7

Table 2.3-1

Reactor Protective System Trip Setting Limits

RPS Segment	Four Reactor Coolant Pumps Operating (Operating Power -100% Rated)	Three Reactor Coolant Pumps Operating (Operating Power -75% Rated)	Two Reactor Coolant Pumps Operating in A Single Loop (Operating Power -46% Rated)	One Reactor Coolant Pump Operating in Each Loop (Operating Power -49% Rated)	Shutdown Bypass
1. Nuclear Power Max. (% Rated)	105.5	105.5	105.5	105.5	5.0(3)
2. Nuclear Power Max. Based on Flow (2) and Imbalance, (% Rated)	1.08 times flow minus reduction due to imbalance	1.08 times flow minus reduction due to imbalance	1.08 times flow minus reduction due to imbalance	1.08 times flow minus reduction due to imbalance	Bypassed
3. Nuclear Power Max. Based on Pump Monitors, (%, Rated)	NA	NA	55% (5)(6)	55%	Bypassed
4. High Reactor Coolant System Pressure, psig, Max.	2355	2355	2355	2355	1720(4)
5. Low Reactor Coolant System Pressure, psig, Min.	1800	1800	1800	1800	Bypassed
6. Variable Low Reactor Coolant System Pressure psig, Min.	(16.25Tout-7769)(1)	(16.25Tout-7769)(1)	(16.25Tout-7769)(1)	(16.25Tout-7769)(1)	Bypassed
7. Reactor Coolant Temp. F., Max.	619	619	619 (6)	619	619
8. High Reactor Building Pressure, psig, Max.	4	4	4	4	4

(1) T_{out} is in degrees Fahrenheit (°F).

(2) Reactor Coolant System Flow, %.

(3) Administratively controlled reduction set only during reactor shutdown.

(4) Automatically set when other segments of the RPS are bypassed.

(5) Reactor power level trip set point produced by pump contact monitor reset to 55.0%.

(6) Specification 3.1.8 applies. Trip one of the two protection channels receiving outlet temperature information from sensors in the idle loop.

Change No. 3

License No. DPR-38

3.1.3 Minimum Conditions for Criticality

Specification

3.1.3.1 The reactor coolant temperature shall be above 525°F except for portions of low power physics testing when the requirements of specification 3.1.9 shall apply.

3.1.3.2 Reactor coolant temperature shall be above DTT + 10°F.

3.1.3.3 When the reactor coolant temperature is below the minimum temperature specified in 3.1.3.1 above, except for portions of low power physics testing when the requirements of specification 3.1.9 shall apply, the reactor shall be subcritical by an amount equal to or greater than the calculated reactivity insertion due to depressurization.

3.1.3.4 The reactor shall be maintained subcritical by at least 1% Δk/k until a steam bubble is formed and a water level between 80 and 396 inches is established in the pressurizer.

3.1.3.5 Except for physics tests and as limited by 3.5.2.1 safety rod groups shall be fully withdrawn prior to any other reduction in shutdown margin by deboration or regulating rod withdrawal during the approach to criticality and before the reactor is critical.

Bases

At the beginning of the initial fuel cycle, the moderator temperature coefficient is expected to be slightly positive at operating temperatures with the operating configuration of control rods.[1] Calculations show that above 525°F, the consequences are acceptable.

Since the moderator temperature coefficient at lower temperatures will be less negative or more positive than at operating temperature,[2] startup and operation of the reactor when reactor coolant temperature is less than 525°F is prohibited except where necessary for low power physics tests.

The potential reactivity insertion due to the moderator pressure coefficient[2] that could result from depressurizing the coolant from 2100 psia to saturation pressure of 900 psia is approximately 0.1% Δk/k.

During physics tests, special operating precautions will be taken. In addition, the strong negative Doppler coefficient[1] and the small integrated Δk/k would limit the magnitude of a power excursion resulting from a reduction of moderator density.

The requirement that the reactor is not to be made critical below DTT + 10°F provides increased assurances that the proper relationship between primary coolant pressure and temperatures will be maintained relative to the NDTT of the primary coolant system. Heatup to this temperature will be accomplished by operating the reactor coolant pumps.

If the shutdown margin required by Specification 3.5.2 is maintained, there is no possibility of an accidental criticality as a result of a decrease of coolant pressure.

The requirement for pressurizer bubble formation and specified water level when the reactor is less than 1% subcritical will assure that the reactor coolant system cannot become solid in the event of a rod withdrawal accident of a start-up accident.[3]

The requirement that the safety rod groups be fully withdrawn before criticality provides an increased shutdown margin during startup.

REFERENCES

(1) FSAR, Section 3

(2) FSAR, Section 3.2.2.1.4

(3) FSAR, Supplement 3, Answer 14.4.1

3.5.2 Control Rod Group and Power Distribution Limits

Applicability

This specification applies to power distribution and operation of control rods during power operation.

Objective

To assure an acceptable core power distribution during power operation, to set a limit on potential reactivity insertion from a hypothetical control rod ejection, and to assure core subcriticality after a reactor trip.

Specification

3.5.2.1 The available shutdown margin shall be not less than 1% $\Delta k/k$ with the highest worth control rod fully withdrawn.

3.5.2.2 Operation with inoperable rods:

a. Operation with more than one inoperable rod, as defined in Specification 4.7.1 and 4.7.2.3, in the safety or regulating rod groups shall not be permitted.

b. If a control rod in the regulating or safety rod groups is declared inoperable in the withdrawn position as defined in Specification 4.7.1.1 and 4.7.1.3, an evaluation shall be initiated immediately to verify the existance of 1% $\Delta k/k$ hot shutdown margin. Boration may be initiated either to the worth of the inoperable rod or until the regulating and transient rod groups are fully withdrawn, whichever occurs first. Simultaneously a program of exercising the remaining regulating and safety rods shall be initiated to verify operability.

c. If within one (1) hour of determination of an inoperable rod as defined in Specification 4.7.1, it is not determined that a 1% $\Delta k/k$ hot shutdown margin exists combining the worth of the inoperable rod with each of the other rods, the reactor shall be brought to the hot standby condition until this margin is established.

d. Following the determination of an inoperable rod as defined in Specification 4.7.1, all rods shall be exercised within 24 hours and exercised weekly until the rod problem is solved.

e. If a control rod in the regulating or safety rod groups is declared inoperable per 4.7.1.2, power shall be reduced to 60% of the thermal power allowable for the reactor coolant pump combination.

f. If a control rod in the regulating or axial power shaping groups is declared inoperable per Specification 4.7.1.2, operation may continue provided the rods in the group are positioned such that the rod that was declared inoperable is maintained within allowable group average position limits of Specification 4.7.1.2.

3.5.2.3 The worth of a single inserted control rod shall not exceed 0.5% $\Delta k/k$ at rated power or 1.0% $\Delta k/k$ at hot zero power except for physics testing when the requirements of Specification 3.1.9 shall apply.

3.5.2.4 Quadrant tilt:

a. If the quadrant power tilt exceeds 5%, except for physics tests, power shall be limited to 90% of the thermal power allowable for the reactor coolant pump combination.

b. If the quadrant power tilt exceeds 10%, except for physics tests, power shall be limited to 80% of the thermal power allowable for the reactor coolant pump combination.

c. If the quadrant power tilt exceeds 20%, except for physics tests, power shall be limited to 60% of the thermal power allowable for the reactor coolant pump combination.

d. Within a period of 4 hours, the quadrant tilt shall be reduced to less than 5%, except for physics tests, or the reactor power/imbalance envelope trip setpoints will be reduced 2% in power for each 1% tilt.

e. If quadrant tilt is in excess of 25%, except for physics tests or diagnostic testing, the reactor will be placed in the hot shutdown condition. Diagnostic testing during power operation with a quadrant power tilt is permitted provided the thermal power allowable for the reactor coolant pump combination is restricted by a reduction of 2% in power for each 1% tilt.

f. Quadrant tilt shall be monitored on a minimum frequency of once every 2 hours during power operation above 15% of rated power.

3.5.2.5 Control rod positions:

a. Technical Specification 3.1.3.5 (safety rod withdrawal) does not prohibit the exercising of individual safety rods as required by Table 4.1-2 or apply to inoperable safety rod limits in Technical Specification 3.5.2.2.

b. Operating rod group overlap shall not exceed 30% between two sequential groups, except for physics tests.

c. Except for physics tests or exercising control rods, the control rod insertion limits are specified on Figure 3.5.2-1 for four pump operation and on Figure 3.5.2-2 for three or two pump operation.

d. If the absolute value of core imbalance is in excess of 5%, except for physics tests, the minimum imbalance achievable shall be determined on a minimum frequency of once per hour until corrected. The axial power shaping rod group shall be used to maintain the core imbalance to within 5% imbalance of the minimum achievable imbalance. The axial power shaping rod group is not restricted from being fully withdrawn.

3.5.2.6 The control rod drive patch panels shall be locked at all times with limited access to be authorized by the superintendent.

<u>Bases</u>

The 30% overlap between sucessive control rod groups is allowed since the worth of a rod is lower at the upper and lower part of the stroke. Control rods are arranged in groups or banks defined as follows:

Group	Function
1	Safety
2	Safety
3	Safety
4	Safety
5	Regulating
6	Regulating
7	Xenon transient override
8	APSR (axial power shaping bank)

Control rod groups are withdrawn in sequence beginning with group 1. Groups 5, 6, and 7 are overlapped 25%. The normal position at power is for groups 6 and 7 to be partially inserted.

The minimum available rod worth provides for achieving hot shutdown by reactor trip at any time assuming, the highest worth control rod remains in the full out position.(1)

Inserted rod groups during power operation will not contain single rod worths greater than 0.5% $\Delta k/k$. This value has been shown to be safe by the safety analysis of the hypothetical rod ejection accident.(2) single inserted control rod worth of 1.0% $\Delta k/k$ at beginning of life, hot, zero power would result in the same transient peak thermal power and therefore the same environmental consequences as a 0.5% $\Delta k/k$ ejected rod worth at rated power.

Date of Issuance: June 8, 1973

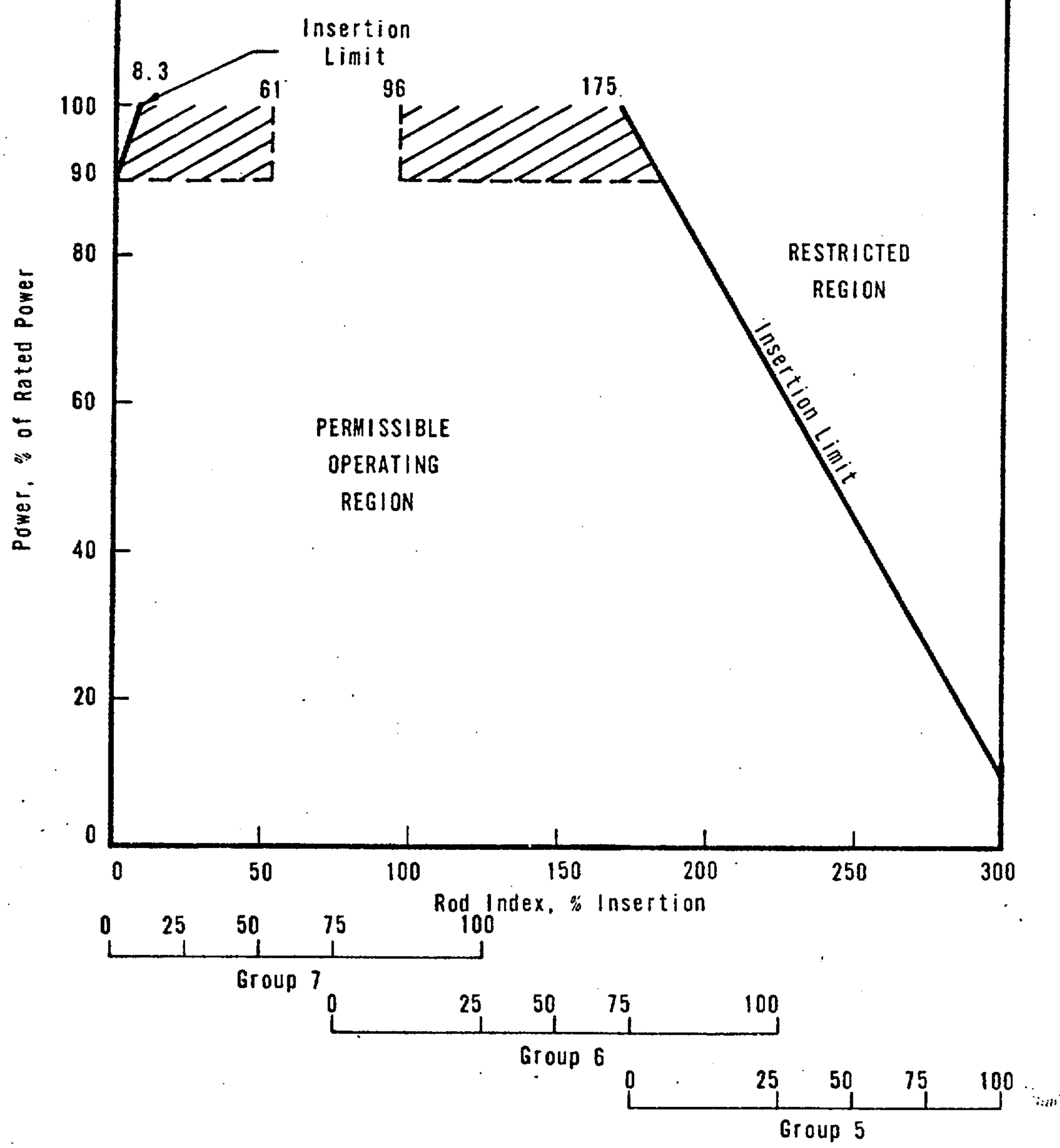
The quadrant power tilt limits set forth in Specification 3.5.2.4 have been established within the thermal analysis design base using the definition of quadrant power tilt given in Technical Specifications, Section 1.6. These limits in conjunction with the control rod position limits in Specification 3.5.2.5c ensure that design peak heat rate criteria are not exceeded during normal operation when including the effects of potential fuel densification.

REFERENCES

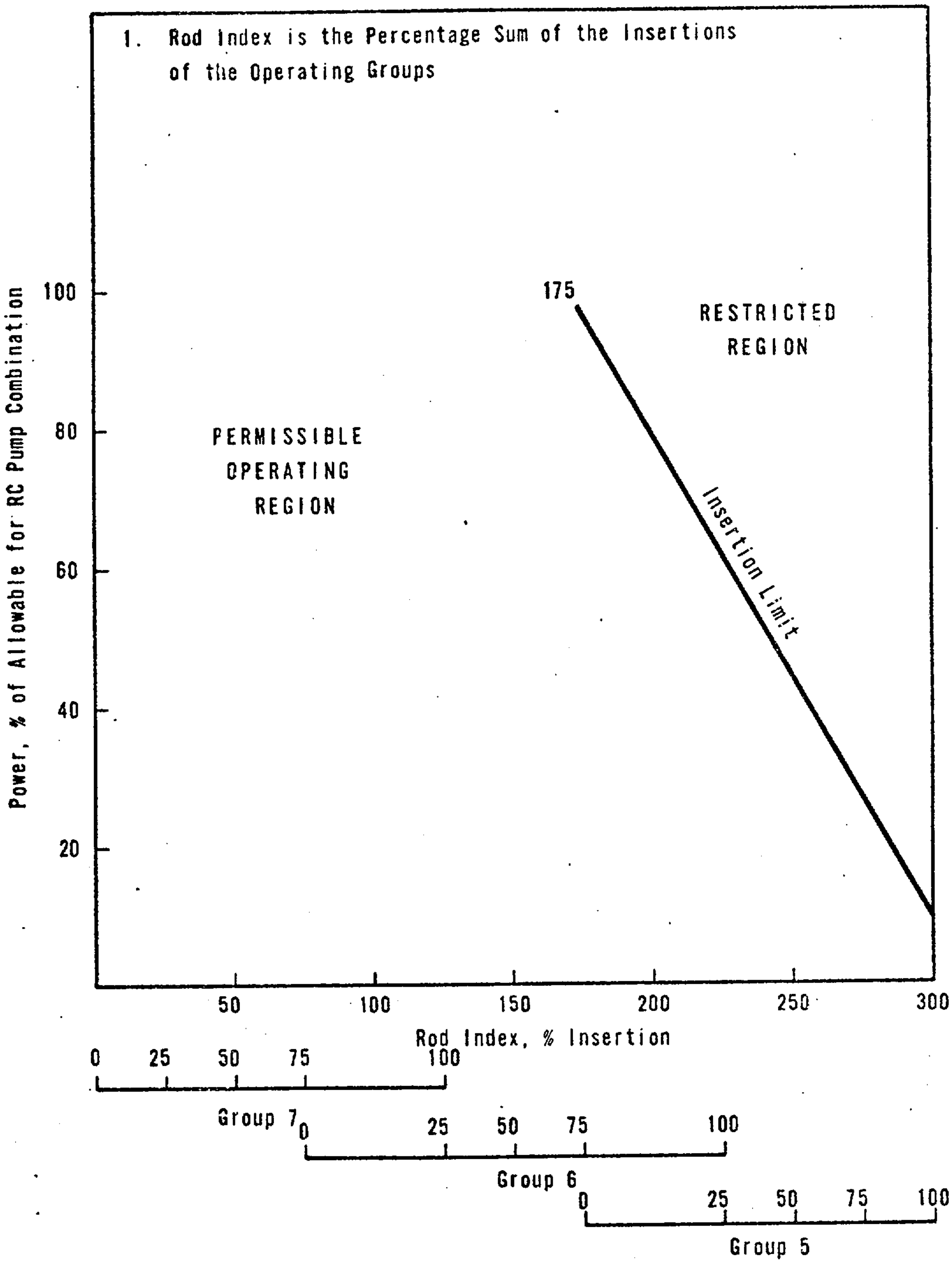
[1]FSAR, Section 3.2.2.1.2

[2]FSAR, Section 14.2.2.2

1. Rod Index is the Percentage Sum of the Insertions of the Operating Groups.
2. The Additional Restrictions on Withdrawal (Hashed Areas) are removed after 100 Full Power Days of operation.



CONTROL ROD GROUP INSERTION LIMITS
FOR 4 PUMP OPERATION



CONTROL ROD GROUP INSERTION LIMITS
FOR 3 AND 2 PUMP OPERATION

OCONEE NUCLEAR STATION

Figure 3.5.2-2

Date of Issuance: June 8 , 1973

3.11 MAXIMUM POWER RESTRICTION

Applicability

Applies to the Nuclear steam supply system of Unit 1 reactor.

Objective

To maintain a power and core life margin in reserve until the system has performed under operating conditions and design objectives for a significant period of time.

Specification

3.11.1 Unit 1 power level may not be increased above 2452 MW_t until operated in the range of 2352 to 2452 MW_t for 30 days, except that 50 percent of the time the power can be as low as 2,000 MW_t, and subsequent approval is granted by the Directorate of Licensing staff.

3.11.2 The first reactor core may not be operated beyond 7500 effective full power hours until supporting analyses and data pertinent to fuel clad collapse under fuel densification conditions have been approved by the Directorate of Licensing staff.

Bases

The Preliminary Safety Analysis Report section of the application for a construction permit was based on a maximum power level of 2452 MW_t. Subsequent safety evaluations done as part of the Final Safety Analysis Report were done for power levels of 2568 MW_t. However, since this is the first nuclear steam supply of this design to go into service, a power margin of 116 MW_t is temporarily being held in reserve until the system has performed at significant power levels for a reasonable period of time. Following evaluation of the summary report of plant startup and power escalation test programs and evaluations, (required by these Technical Specifications), and in the absence of any significant deviation in plant performance from that predicted by design and required for safety, it is expected that this temporary restriction will be lifted.

The Licensing staff has reviewed the effects of fuel densification for the first core in Oconee Unit 1 and concluded that clad collapse will not take place within the first fuel cycle (7500 effective full power hours). However, the clad collapse model used is questionable for extrapolation of clad collapse time out beyond the first fuel cycle because of limited experimental verification.

Date of Issuance: June 3 , 1973

4.0 SURVEILLANCE STANDARDS

Specified intervals may be adjusted plus or minus 25% to accommodate normal test schedules.

4.1 OPERATIONAL SAFETY REVIEW

Applicability

Applies to items directly related to safety limits and limiting conditions for operation.

Objective

To specify the minimum frequency and type of surveillance to be applied to unit equipment and conditions.

Specification

4.1.1 The minimum frequency and type of surveillance required for reactor protective system and engineered safety feature protective system instrumentation when the reactor is critical shall be as stated in Table 4.1-1.

4.1.2 Equipment and sampling test shall be performed as detailed in Tables 4.1-2 and 4.1-3.

4.1.3 When the reactor is above 95% of full rated power, a power distribution map shall be made at least weekly using the incore instrumentation detector system. The results shall be within the acceptance criteria of an approved procedure or the power shall be reduced to less than 95% of full rated power until corrected.

Bases

Check

Failures such as blown instrument fuses, defective indicators, faulted amplifiers which result in "upscale" or "downscale" indication can be easily recognized by simple observation of the functioning of an instrument or system. Furthermore, such failures are, in many cases, revealed by alarm or annuniciator action. Comparison of output and/or state of independent channels measuring the same variable supplements this type of built-in surveillance. Based on experience in operation of both conventional and nuclear systems, when the unit is in operation, the minimum checking frequency stated is deemed adequate for reactor system instrumentation.

Date of Issuance: June 3 , 1973

Calibration

Calibration shall be performed to assure the presentation and acquisition of accurate information. The nuclear flux (power range) channels amplifiers shall be calibrated (during steady state operating conditions) when indicated neutron power and core thermal power differ by more than 2 percent. During non-steady state operation, the nuclear flux channels amplifiers shall be calibrated daily to compensate for instrumentation drift and changing rod patterns and core physics parameters.

Channels subject only to "drift" errors induced within the instrumentation itself can tolerate longer intervals between calibrations. Process system instrumentation errors induced by drift can be expected to remain within acceptable tolerances if recalibration is performed at the intervals of each refueling period.

Substantial calibration shifts within a channel (essentially a channel failure) will be revealed during routine checking and testing procedures.

Thus, minimum calibration frequencies set forth are considered acceptable.

Testing

On-line testing of reactor protective channels is required once every four weeks on a rotational or perfectly staggered basis. The rotation scheme is designed to reduce the probability of an undetected failure existing within the system and to minimize the likelihood of the same systematic test errors being introduced into each redundant channel.

The rotation schedule for the reactor protective channels is as follows:

Channels A, B, C & D	Before Startup
Channel A	One Week After Startup
Channel B	Two Weeks After Startup
Channel C	Three Weeks After Startup
Channel D	Four Weeks After Startup

The reactor protective system instrumentation test cycle is continued with one channel's instrumentation tested each week. Upon detection of a failure that prevents trip action, all instrumentation associated with the protective channels will be tested after which the rotational test cycle is started again. If actuation of a safety channel occurs, assurance will be required that actuation was within the limiting safety system setting.

Date of Issuance: June 8 , 1973

The protective channels coincidence logic and control rod drive trip breakers are trip tested every four weeks. The trip test checks all logic combinations and is to be performed on a rotational basis. The logic and breakers of the four protective channels shall be trip tested prior to startup and their individual channels trip tested on a cyclic basis. Discovery of an unsafe failure requires the testing of all channel logic and breakers, after which the trip test cycle is started again.

The equipment testing and system sampling frequencies specified in Table 4.1-2 and Table 4.1-3 are considered adequate to maintain the equipment and systems in a safe operational status.

Power Distribution Mapping

The incore instrumentation detector system will provide a means of assuring that axial and radial power peaks and the peak locations are being controlled by the provisions of the Technical Specifications within the limits employed in the safety analysis.

REFERENCE

FSAR, Section 7.1.2.3.4

4.7 REACTOR CONTROL ROD SYSTEM TESTS

4.7.1 Control Rod Drive System Functional Tests

Applicability

Applies to the surveillance of the control rod system.

Objective

To assure operability of the control rod system.

Specification

4.7.1.1 The control rod trip insertion time shall be measured for each control rod at either full flow or no flow conditions following each refueling outage prior to return to power. The maximum control rod trip insertion time for an operable control rod drive mechanism, except for the Axial Power Shaping Rods (APSRs), from the fully withdrawn position to 3/4 insertion (104 inches travel) shall not exceed 1.66 seconds at reactor coolant full flow conditions or 1.40 seconds for no flow conditions. For the APSRs it shall be demonstrated that loss of power will not cause rod movement. If the trip insertion time above is not met, the rod shall be declared inoperable.

4.7.1.2 If a control rod is misaligned with its group average by more than an indicated nine (9) inches, the rod shall be declared inoperable and the limits of Specification 3.5.2.2 shall apply. The rod with the greatest misalignment shall be evaluated first. The position of a rod declared inoperable due to misalignment shall not be included in computing the average position of the group for determining the operability of rods with lesser misalignments.

4.7.1.3 If a control rod cannot be exercised, or if it cannot be located with absolute or relative position indications or in or out limit lights, the rod shall be declared to be inoperable.

Bases

The control rod trip insertion time is the total elapsed time from power interruption at the control rod drive breakers until the control rod has completed 104 inches of travel from the fully withdrawn position. The specified trip time is based upon the safety analysis in FSAR, Section 14.

Each control rod drive mechanism shall be exercised by a movement of approximately two (2) inches of travel every two (2) weeks. This requirement shall apply to either a partial or fully withdrawn control rod at reactor operating conditions. Exercising the drive mechanisms in this manner provides assurance of reliability of the mechanisms.

A rod is considered inoperable if it cannot be exercised, if the trip insertion time is greater than the specified allowable time, or if the rod deviates from its group average position by more than nine (9) inches. Conditions for operation with an inoperable rod are specified in Technical Specification 3.5.2[(2)]

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Bases

The control rod trip insertion time is the total elapsed time from power interruption at the control rod drive breakers until the control rod has completed 104 inches of travel from the fully withdrawn position. The specified trip time is based upon the safety analysis in FSAR, Section 14.

Each control rod drive mechanism shall be exercised by a movement of approximately two (2) inches of travel every two (2) weeks. This requirement shall apply to either a partial or fully withdrawn control rod at reactor operating conditions. Exercising the drive mechanisms in this manner provides assurance of reliability of the mechanisms.

A rod is considered inoperable if it cannot be exercised, if the trip insertion time is greater than the specified allowable time, or if the rod deviates from its group average position by more than nine (9) inches. Conditions for operation with an inoperable rod are specified in Technical Specification 3.5.2[(2)]

REFERENCES

(1) FSAR, Section 14

(2) Technical Specification 3.5.2

Date of Issuance: June 8 , 1973

UNITED STATES ATOMIC ENERGY COMMISSION

SAFETY EVALUATION BY THE DIRECTORATE OF LICENSING

DOCKET NO. 50-269

OCONEE NUCLEAR STATION UNIT 1

INTRODUCTION

The Oconee Unit 1 power plant was licensed for full power operation on February 6, 1973. The operation of the plant has been restricted by Technical Specifications to 75% of full rated power pending completion of the Regulatory staff's fuel densification evaluation. The fuel densification evaluation has been completed and by letter dated May 14, 1973, Duke Power Company (the licensee) requested that the power restriction be rescinded.

EVALUATION

General

The staff reviewed the effects of fuel densification for Oconee 1 on the basis of the staff's guidelines and the technical evaluation of the applicant's safety analysis of steady state operation, operating transients and postulated accidents. In the evaluation the applicant appropriately considered the staff guidelines including the effects of instantaneous and anisotropic densification (initial density minus 2σ, final density 96.5% TD), the assumption of no clad creepdown as a function of core life, and the assumption of an axial gap leading to a power spike. The staff reviewed the effects of fuel manufacturing and reactor operating parameters on the fuel densification mechanism. The staff reviewed the applicant's assumptions, methods, and computer codes used in evaluating the fuel densification effects. The mechanical integrity of the fuel cladding and thermal performance of the fuel were considered in the analyses of steady state operation, operating transients, and postulated accidents as discussed in the following sections.

Mechanical Integrity of Cladding

Clad creepdown during the core life is not considered by the applicant in the calculation of gap conductance. This is a conservative assumption since the reduced gap size due to clad creepdown would result in a higher gap conductance and thus in a lower stored energy in the fuel. The staff reviewed the B&W method for calculating the clad collapse time, which is the time required for an unsupported clad tubing to flatten into the axial gap volume caused by fuel densification. On the basis of independent staff calculations and from experience of fuel performance in other reactors, the staff concurs with the applicant that clad collapse is not expected for the Oconee 1 fuel during the first cycle of 7500 effective full power hours (EFPH). However, the staff concludes that the evaluation model for collapse time calculations contains several deficiencies in its application to Oconee 1. The staff has informed the applicant that a final resolution of the B&W model for collapse time calculations is necessary for subsequent fuel cycles of Oconee 1.

Fuel Pin Thermal Analysis

The applicant uses the B&W computer code, TAFY, to calculate gap conductance, fuel temperature, and stored energy for the Oconee 1 fuel, used as a basis for the safety analysis. To demonstrate the applicability of the TAFY code for the evaluation of the Oconee 1 fuel thermal behavior, the applicant compared TAFY predicted fuel temperatures and gap conductances with experimental data.

The staff reviewed the TAFY code and concludes that realistic and/or conservative assumptions have been used for the modeling of the physical phenomena incorporated into the code (thermal expansion, fuel swelling, sorbed gas release, fission gas release), with two exceptions: (1) partial contact between the clad and fuel and (2) formation of a central void due to fuel restructuring on the basis of columnar grain growth at a temperature of 3200°F. These assumptions are discussed below.

The assumption of a partial contact between fuel and clad is based on the work by Kjaerheim and Rolstad who determined the UO_2 thermal conductivity from measured fuel temperatures using a fuel clad geometry with cold diametral gaps ranging from 1.85 mil to 6.61 mil. In order to predict measured temperatures Kjaerheim and Rolstad developed an analytical model which assumes heat transfer not only by radiation and conduction through the gap but also assumes conductive heat transfer at a partial contact area, CA, between fuel and clad (conduction through gas in contact area and conduction at solid-to-solid interface), which is attributed to fuel cracking. The CA model predicts a minimum contact area of 10 percent regardless of the initial diametral gap for a particular fuel diameter. For the Oconee 1 fuel-clad geometry with a cold diametral gap of 12.8 mil the partial contact area is 11 percent, on the average. The TAFY calculated gap conductance for Oconee 1 at BOL and with a linear heat rate of 16 kW/ft is 1052 Btu/hr-ft^2-F° of which approximately 1/3 is due to heat transfer across the partial contact area. The comparison of TAFY predicted temperatures and gap conductances with corresponding experimental values shows that TAFY is conservative for small diametral gaps ($\leq$ 6.61 mil), but for gap sizes comparable to the Oconee 1 gap, and larger, the code predictions are not consistent and can be either conservative or not conservative. The staff concludes that the use of the partial contact area model may be very useful for predicting temperatures of an experiment from which the model was derived, but should not be used when disassociated from other features in the model and when extrapolated to other gap sizes.

The TAFY code uses a fuel restructuring model based on the assumption of columnar grain growth at 3200°F with a change in fuel density from 96.5% TD which leads to the formation of a central void in the fuel, and results in a reduction of maximum fuel temperature and stored energy. The staff reviewed this assumption on the basis of photomicrographs of cross sections of exposed typical B&W fuel, and concludes that fuel restructuring associated with the formation of a central void can take place under certain operating histories and conditions. However, these parameters are not necessarily known for the Oconee 1 reactor operating conditions and it would be possible that the irradiation induced fuel densification has been completed before a temperature of 3200° is achieved and, therefore the possibility of fuel restructuring due to columnar grain growth would be foreclosed.

CONCLUSION

The staff has concluded that there is a reasonable assurance that fuel clad collapse will not take place during the first fuel cycle (7500 EFPH) of the Oconee Unit 1. The staff has informed the applicant that the method to calculate clad collapse time is not acceptable and the applicant has committed to develop a model acceptable to the staff prior to a second fuel cycle of the Oconee Unit 1.

The staff has concluded that the assumptions in the TAFY code (1) of partial contact between fuel and cladding and (2) fuel restructuring leading to a central void in the fuel has not been justified by B&W and are not acceptable. In absence of a timely resolution of the two assumptions for Oconee Unit 1, the staff requested the applicant to remove the assumption of restructuring from the TAFY code and to reduce the TAFY calculated gap conductance by 25%. With this reduction TAFY predictions for experimental data will be conservative for gap sizes comparable to Oconee 1. The staff concludes that the use of as-built dimensions for the fuel and clad in the TAFY code is acceptable with the exception that the initial fuel density with a minus 2σ value is to be assumed, consistent with the staff fuel densification report. The applicant has recalculated gap conductance and fuel temperatures with these assumptions and used these values to establish maximum linear heat rates. In order to prevent fuel melting at a temperature of 5080°F, the maximum allowable linear heat rate is calculated to be 20.1kW/ft and in order to not exceed a clad temperature of 2300°F (ECCS criteria) a maximum linear heat rate of 18.6 kW/ft has been calculated.

I. A. Peltier

I. A. Peltier, Project Manager
Pressurized Water Reactors Branch No. 4
Directorate of Licensing

A. Schwencer

A. Schwencer, Chief
Pressurized Water Reactors Branch No. 4
Directorate of Licensing