5/22/73

Docket No. 50-266

Wisconsin Electric Power Company
Wisconsin Michigan Power Company
ATTN: Mr. Sol Burstein
Senior Vice President
231 West Michigan Street
Milwaukee, Wisconsin

Technical Specification Change No. 4
License No. DPR-24

Gentlemen:

Your letter dated February 5, 1973 to Mr. A. Giambusso requested our evaluation of your proposed operation of Cycle 2 for Unit 1 of the Point Beach Nuclear Plant. In support of this proposal you submitted an analysis of the effects of fuel densification on Cycle 2 operation of Point Beach Nuclear Plant Unit 1 (Westinghouse report WCAP-8051, non-proprietary) on February 5, 1973. The analysis follows the methods previously presented for Point Beach Unit 2 and reviewed by the staff, and in addition takes into account expected fuel clad flattening in regions 2 and 3 of the Cycle 2 core. You have concluded from the analysis that full power operation to a fuel exposure of 8000 effective full power hours (EFPH) is justified, with appropriate provisions made for reduced power peaking and for limiting steam generator leakage.

In our letter to you dated March 2, 1973, we concluded that interim operation of Unit 1 at a power level no greater than 75% of rated full power in accordance with special operating limitations imposed during the latter part of Cycle 1 (enclosed as an attachment to our letter of August 23, 1972) would provide an acceptable margin of safety pending completion of our review.

We have now completed our review of the report "Fuel Densification, Point Beach Nuclear Plant Unit 1 - Cycle 2," dated January 1973. In order to assure for safe operation with collapsed fuel rods you have:

(1) limited the clad temperature in collapsed sections of a fuel rod to less than 1800°F during a loss-of-coolant accident.

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(2) Adequately allowed for the large power spike that will result in rods adjacent to collapsed rods.

(3) Included an additional 1.9% decrease in DNBR to account for increased pellet clad eccentricity and reduced fuel rod circumference and heat transfer area.

(4) Included a 10% penalty applied at the point of minimum DNBR to conservatively account for possible contact of rods due to flattening and bowing.

(5) Changed the reference axial power distribution for DNB analysis from a chopped cosine with a 1.72 peak to average power to one with a 1.55 peak to average power.

(6) Reduced the overall peaking factor F_Q from 2.80 to 2.60 to allow for local power peaking due to fuel densification and flattened cladding.

On the basis of the above and our review of your report we have determined that the effects of fuel densification have been adequately analyzed and that the plant can be operated at 100% of rated power with appropriate changes to the Technical Specifications.

Changes to the Technical Specifications have been made in Section 2.1 to limit fuel residence time to 8000 EFPH; to Section 2.3 to reduce the high flux set point to 108% of rated power and to revise the overtemperature constants; to Section 3.1-D to limit primary to secondary steam generator leakage so that in the event of an overpower transient and failure of all flattened fuel rods the radiation exposure at the site boundary will not exceed the limits of 10 CFR 20; and to Section 3.10 to incorporate revised full length and part length control rod insertion limits, power distribution limits, and additional power distribution surveillance requirements.

Your letter dated October 17, 1972 requested authorization to change the Point Beach Nuclear Plant Unit No. 1 Technical Specifications to allow operation with one Westinghouse test fuel assembly as a part of Region 4. In support of this request you provided a document entitled "Description and Safety Analysis for Proposed Change No. 4 to Technical Specifications."

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This test fuel assembly is basically similar to the standard fuel assembly but provides for easy removal and inspection of 51 of the fuel rods. Twelve of the rods use a different Zircaloy tubing to encapsulate the fuel pellets. Our review of your submittal indicates that:

(1) The test assembly has been designed to the same criteria as a standard fuel assembly with no significant effect on core performance. Similar test assemblies have been operated in other power reactors.

(2) The fuel rod clad performance for the 12 rods using different Zircaloy tubing is expected to be equal or superior to the reference Zircaloy-4 cladding.

(3) Test fuel rods in the test assembly have been designed within conservative limits and can be expected to have no effect on safety during operation.

(4) The test assembly will have no effect on in-core performance under LOCA conditions.

On the basis of the above considerations we conclude that inclusion of the test fuel assembly will not affect the safe operation of the reactor core and that Section 15.5.3.A.1 can be changed as per the enclosed page 15.5.3-1 to include the test fuel assembly.

We conclude that all the changes to the Technical Specifications described above do not involve significant hazard considerations not described or implicit in the Safety Analysis Report and 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, the Technical Specifications of Facility Operating License No. DPR-34 are hereby changed as set forth in revised Sections 2.1, 2.3, 3.1-D, 3.10, and 5.1, and designated Change No. 4, copies of which are enclosed. Although every page of every section has not been changed, the entire sections with their bases are being replaced as separate entities. Portions of the

Specifications that have been changed are designated by a numeral 4 in the right hand margin.

Sincerely,

R. C. DeYoung, Assistant Director
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Form AEC-318 (Rev. 9-53) AECM 0240 GPO c43—16—81465-1 445-678

15.5.3 REACTOR

Applicability

Applies to the reactor core, Reactor Coolant System, and Emergency Core Cooling Systems.

Objective

To define those design features which are essential in providing for safe system operation.

Specifications

A. Reactor Core

1. The reactor core contains approximately 48 metric tons of uranium in the form of slightly enriched uranium dioxide pellets. The pellets are encapsulated in Zircaloy-4 tubing to form fuel rods. The reactor core is made up of 121 fuel assemblies. Each fuel assembly contains 179 fuel rods.(1) One of the reload fuel assemblies for Unit 1 Cycle 2 will be a special assembly containing 51 removable fuel rods, up to twelve (12) of which have a different zirconium alloy tubing to encapsulate the fuel pellets. This assembly will remain 4 in the reactor core through three normal reactor core cycles. This assembly will not be placed in a control rod position at any time during its irradiation.

2. The average enrichment of the initial core is a nominal 2.90 weight percent of U-235. Three fuel enrichments are used in the initial core. The highest enrichment is a nominal 3.40 weight percent of U-235.(2)

3. Standard reload fuel will be similar in design to the initial core.

4. Burnable poison rods are incorporated in the initial core. There are 704 poison rods in the form of 8, 12 and 16 rod clusters, which are located in vacant rod cluster control guide tubes.(3) The burnable poison rods consist of borated pyrex glass clad with stainless steel.(4)

5. There are 33 full-length RCC assemblies and 4 partial-length RCC assemblies in the reactor core. The full-length RCC assemblies contain a 142 inch length of silver-indium-cadmium alloy clad with the stainless steel. The partial-length RCC assemblies contain a 36 inch length of silver-indium-cadmium alloy with the remainder of the stainless steel sheath filled with Al_2O_3.(5)

6. Up to ten (10) grams of enriched fissionable material may be used either in the core, or available on the plant site, in the form of fabricated neutron flux detectors for the purposes of monitoring core neutron flux.

B. Reactor Coolant System

1. The design of the Reactor Coolant System complies with the code requirements.(6)

2. All high pressure piping, components of the Reactor Coolant System and their supporting structures are designed to Class I requirements, and have been designed to withstand:

   a. The design seismic ground acceleration, 0.06g, acting in the horizontal and 0.04g acting in the vertical planes simultaneously, with stresses maintained within code allowable working stresses.

b. The maximum potential seismic ground acceleration, 0.12g, acting in the horizontal and 0.08g acting in the vertical planes simultaneously with no loss of function.

3. The nominal liquid volume of the Reactor Coolant System, at rated operating conditions, is 6040 cubic feet.

References

(1) FSAR Section 3.2.3

(2) FSAR Section 3.2.1

(3) FSAR Section 3.2.1

(4) FSAR Section 3.2.3

(5) FSAR Sections 3.2.1 & 3.2.3

(6) FSAR Table 4.1-9

CHANGE NO. 4 TO
TECHNICAL SPECIFICATIONS FOR
FACILITY OPERATING LICENSE NO. DPR-24
FOR POINT BEACH NUCLEAR PLANT UNIT 1
WISCONSIN ELECTRIC POWER CO.
WISCONSIN MICHIGAN POWER CO.
DOCKET NO. 50-266

15.2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

15.2.1 SAFETY LIMIT, REACTOR CORE

Applicability:

Applies to the limiting combinations of thermal power, reactor coolant system pressure, and coolant temperature during operation.

Objective:

To maintain the integrity of the fuel cladding

Specification:

1. The combination of thermal power level, coolant pressure, and coolant temperature shall not exceed the limits shown in Figure 15.2.1-1. The safety limit is exceeded if the point defined by the combination of reactor coolant system average temperature and power level is at any time above the appropriate pressure line.
2. The fuel residence time for Unit 1, Cycle 2, shall be
   presently limited to 8,000 effective full power hours
   (EFPH) under design operating conditions. The Licensee 4
   may propose to operate the core in excess of 8,000 EFPH
   by providing an analysis which includes the effect of
   further clad flattening or a change in operating
   conditions. Any such analysis, if proposed, shall be
   approved by the Regulatory staff prior to operation
   in excess of 8,000 EFPH.

Basis:

To maintain the integrity of the fuel cladding and prevent fission product release, it is necessary to prevent overheating of the cladding under all operating conditions. This is accomplished by operating the hot regions of the core within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is vary large and the clad surface temperature is only a few degrees Fahrenheit above the coolant saturation temperature. The upper boundary of the nucleate boiling regime is termed departure from nucleate boiling (DNB) and at this point there is a sharp reduction of the heat transfer coefficient, which would result in high clad temperatures and the possibility of clad failure. DNB is not, however, an observable parameter during reactor operation. Therefore, the observable parameters; thermal power, reactor coolant temperature and pressure have been related to DNB through the W-3 DNB correlation. The W-3 DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB. The minimum value of the DNB ratio, DNBR, during steady state operation, normal operational transients, and anticipated transients is limited to 1.30. A DNB ratio of 1.30 corresponds to a 95% probability at a 95% confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.

The curves of Figure 15.2.1-1 represent the loci of points of thermal power, coolant system pressure and average temperature for which the DNB ratio is no less than 1.30. The area of safe operation is below these lines. The safety limits curves have been revised
to allow for heat flux peaking effects due to fuel densification 4
and flattened fuel cladding sections.

Additional peaking factors to account for local peaking due to fuel rod axial gaps and reduction in fuel pellet stack length as well as a penalty to account for rod bowing, have been included in the calculation of the curves shown in Figure 15.2.

These curves are based on an $F^N_{\Delta H}$ of 1.58, a 1.55 cosine axial flux shape, and a DNB analysis as described in Section 4.3 of WCAP-8050 "Fuel Denficiation, Point Beach Nuclear Plant Unit 1 Cycle 2," (including the effects of fuel densification and flattened cladding). 4

Figure 15.2.1-1 also includes an allowance for an increase in the enthalpy rise hot channel factor at reduced power based on the expression:

$F^N_{\Delta H}$ = 1.58 [1 + 0.2 (1-P)] where P is the fraction of rated power when $P \leq 1.0$. $F^N_{\Delta H}$ = 1.58 when $P > 1.0$.

The hot channel factors are also sufficiently large to account for the degree of malpositioning of full-length rods that is allowed before the reactor trip set points are reduced and rod withdrawal block and load runback may be required. Rod withdrawal block and load runback occur before reactor trip setpoints are reached.

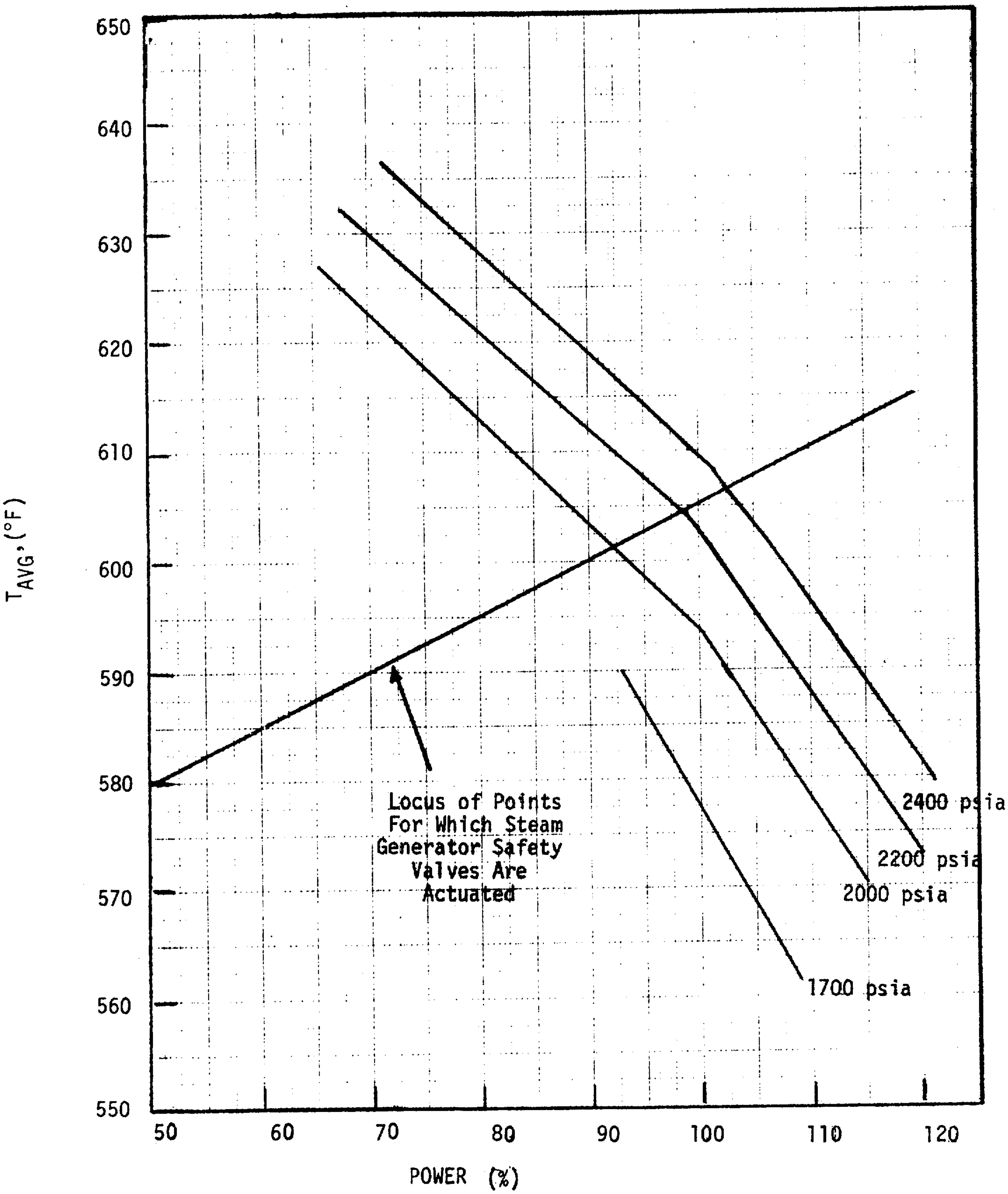
The Reactor Control and Protective System is designed to prevent any anticipated combination of transient conditions that would result in a DNB ratio of less than 1.30.

The fuel residence time for Unit 1, Cycle 2 is limited to 8.000 EFPH to assure no fuel clad flattening without prior review by the Regulatory staff. Prior to 8,000 EFPH, the licensee may provde the additional analyses required for operation beyond 8,000 EFPH. 4

FIGURE 15.2.1-1

CORE DNB SAFETY LIMITS

POINT BEACH 1 - CYCLE 2



15.2.3 LIMITING SAFETY SYSTEM SETTINGS, PROTECTIVE INSTRUMENTATION

Applicability:

Applies to trip settings for instruments monitoring reactor power and reactor coolant pressure, temperature, flow, pressurizer level, and permissives related to reactor protection.

Objective:

To provide for automatic protective action in the event that the principal process variables approach a safety limit.

Specification:

1. Protective instrumentation for reactor trip settings shall be as follows:

   A. Startup protection

      (1) High flux, source range - within span of source range instrumentation.

      (2) High flux, intermediate range - $\leq$ 40% of rated power.

      (3) High flux, power range (low set point) - $\leq$ 25% of rated power.

   B. Core limit protection

      (1) High flux, power range (high setpoint)
      $\leq$ 108% of rated power 4

      (2) High pressurizer pressure - $\leq$ 2385 psig.

(3) Low pressurizer pressure - $\geq$ 1875 psig.

(4) Overtemperature ΔT

$$\leq \Delta T_o \left[K_1 - K_2 (T-T') \left(\frac{1+\tau_1 S}{1+\tau_2 S}\right) + K_3 (P-P') - f(\Delta I)\right]$$

where

ΔT_o = indicated ΔT at rated power, °F

T = average temperature, °F

T' = 581.3°F

P = pressurizer pressure, psig

P' = 2235 psig

K_1 = 1.037

K_2 = 0.00948

K_3 = 0.000527 4

τ_1 = 25 sec

τ_2 = 3 sec

and $f(\Delta I)$ is an even function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests, where q_t and q_b are the percent power in the top and bottom halves of the core respectively, and $q_t + q_b$ is total core power in percent of rated power, such that:

(a) for $q_t - q_b$ within -17, +9 percent, $f(\Delta I) = 0$.

(b) for each percent that the magnitude of $q_t - q_b$ exceeds +9
percent the ΔT trip set point shall be automatically reduced 4
by an equivalent of four percent of rated power.

(c) for each percent that the magnitude of $q_t - q_b$ exceeds -17 percent the ΔT trip setpoint shall be automatically reduced by an equivalent of 2.5 percent of rated power. 4

[1.B. (5)] Overpower ΔT

$$\leq \Delta T_o \left[K_4 - K_5 \frac{\tau_3 S}{\tau_3 S + 1} T - K_6 (T-T') - f(\Delta I)\right]$$

where

ΔT_o = indicated ΔT at rated power, °F

T = average temperature, °F

T' = 581.3°F

K_4 $\leq$ 1.08 of rated power 4

K_5 = 0.0262 for increasing T

= 0.0 for decreasing T

K_6 = 0.0 for $T<T'$

= 0.002 for $T \geq T'$

τ_3 = 10 sec

f (ΔI) as defined in (4) above,

(6) Undervoltage - $\geq$ 75% of normal voltage

(7) Low indicated reactor coolant flow per loop- $\geq$90% of normal indicated loop flow

(8) Reactor coolant pump motor breaker open

(a) Low frequency set point $\geq$ 57.5 cps

(b) Low voltage set point $\geq$ 75% of normal voltage

C. Other reactor trips

(1) High pressurizer water level - $\leq$ 95% of span

(2) Low-low steam generator water level - $\geq$5% of narrow range instrument span

(3) Steam-Feedwater Flow Mismatch Trip - $\leq 1.0 \times 10^6$ lb/hr

(4) Turbine Trip (Not a protection circuit)

(5) Safety Injection Signal

(6) Manual Trip

2. Protective instrumentation settings for reactor trip interlocks shall be as follows:

A. The "at power" reactor trips (low pressurizer pressure, high pressurizer level, and low reactor coolant flow for both loops) shall be unblocked when:

(1) Power range nuclear flux $\geq$ 10% of rated power or,

(2) Turbine Load $\geq$ 10% of full load turbine pressure. 4

B. The single loss of flow trip shall be unblocked when the power range nuclear flux $\geq$ 50% of rated power.

C. The power range high flux level low range trip, and intermediate range high flux level trip shall be unblocked when power is $\leq$ 10% of rated power.

D. The source range high flux reactor trip shall be unblocked when the intermediate range flux is $\leq 10^{-10}$ amperes.

Basis

The source range high flux reactor trip prevents a startup accident from subcritical conditions from proceeding into the power range. Any set point within its range would prevent an excursion from proceeding to the point at which significant thermal power is generated.[1]

The high flux low power reactor trip provides redundant protection in the power range for a power excursion beginning from low power. This trip insures that a more restrictive trip point is used for this case than for an excursion beginning from near full power.[1]

The overpower nuclear flux reactor trip protects the reactor core against reactivity excursions which are too rapid to be protected by temperature and pressure circuitry. The prescribed set point, with allowance for errors, is consistent with the trip point assumed in the accident analysis.[3]

The overpower ΔT reactor trip prevents power density anywhere in the core from exceeding 108% of design power density, and includes corrections for axial power distribution, change in density and heat capacity of water with temperature, and dynamic compensation for piping delays from the core to the loop temperature detectors. The specified set points meet this requirement and include allowance for instrument errors.[2]

The overtemperature ΔT reactor trip provides core protection against DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided only that (1) the transient is slow with respect to piping transit delays from the core to the temperature detectors, (about 4 seconds),[5] and (2) pressure is within the range between the high and low pressure reactor trips. With normal axial power distribution, the reactor trip limit, with allowance for errors,[2] is always below the core safety limit as shown on Figure 15.2.1-1. If axial peaks are greater than design, as indicated by difference between top and bottom power range nuclear detectors, the reactor trip limit is automatically reduced.[6] [7]

The overpower and overtemperature protections system setpoints have been revised to include effects of fuel densification and clad flattening on core safety limits. The revised setpoints as given above will ensure that the combination of power, temperature, and pressure will not exceed the revised core safety limits as shown in Figure 15.2.1-1 4

The overpower limit criteria is that core power be prevented from reaching a value at which fuel pellet centerline melting would occur. The reactor is prevented from reaching the overpower limit condition by action of the nuclear overpower and overpower ΔT trips.

The high and low pressure reactor trips limit the pressure range in which reactor operation is permitted. The high pressurizer pressure reactor trip setting is lower than the set pressure for the safety valves (2485 psig) such that the reactor is tripped before the safety valves actuate. The low pressurizer pressure reactor trip trips the reactor in the unlikely event of a loss-of-coolant accident.[4]

The low flow reactor trip protects the core against DNB in the event of either a decreasing actual measured flow in the loops or a sudden loss of power to one or both reactor coolant pumps. The set point specified is consistent with the value used in the accident analysis.[8] The low loop flow signal is caused by a condition of less than 90% flow as measured by the loop flow instrumentation. The loss of power signal is caused by the reactor coolant pump breaker opening as actuated by either high current, low supply voltage or low electrical frequency, or by a manual control switch. The significant feature of the breaker trip is the frequency set-point, $\geq$57.5 cps, which assures a trip signal before the pump inertia is reduced to an unacceptable value.

The high pressurizer water level reactor trip protects the pressurizer safety valves against water relief. The specified set point allows adequate operating instrument error[2] and transient overshoot in level before the reactor trips.

The low-low steam generator water level reactor trip protects against loss of feedwater flow accidents. The specified set point assures that there will be sufficient water inventory in the steam generators at the time of trip to allow for starting delays for the auxiliary feedwater system.[9]

Numerous reactor trips are blocked at low power where they are not required for protection and would otherwise interfere with normal plant operations. The prescribed set point above which these trips are unblocked assures their availability in the power range where needed.

Sustained operation with only one pump will not be permitted above 10% power. If a pump is lost while operating between 10% and 50% power, an orderly and immediate reduction in power level to below 10% is allowed. The power-to-flow ratio will be maintained equal to or less than unity, which ensures that the minimum DNB ratio increases at lower flow because the maximum enthalpy rise does not increase above the maximum enthalpy rise which occurs during full power and full flow operation.

References

(1) FSAR 14.1.1

(2) FSAR, page 14-3

(3) FSAR 14.2.6

(4) FSAR 14.3.1

(5) FSAR 14.1.2

(6) FSAR 7.2, 7.3

(7) FSAR 3.2.1

(8) FSAR 14.1.9

(9) FSAR 14.1.11

D. LEAKAGE OF REACTOR COOLANT

Specification:

1. If leakage of reactor coolant is indicated to exceed 1 GPM by the means available such as water inventory balances, monitoring equipment or direct observation, a follow-up evaluation of the safety implications shall be initiated as soon as practicable but no later than within 4 hours. Any indicated leak shall be considered to be a real leak until it is determined that either (1) a safety problem does not exist or (2) that the indicated leak cannot be substantiated by direct observation or other indication. 2

2. If the indicated leakage is substantiated and is not evaluated as safe or is determined to exceed 10 GPM, reactor shutdown shall be initiated as soon as practicable but no later than within 24 hours after the leak was first detected.

3. The nature of the leak as well as the magnitude of the leak shall be considered in the safety evaluation. If plant shutdown is necessary per specification 2 above, the rate of shutdown and the conditions of shutdown shall be determined by the safety evaluation for each case and justified in writing as soon thereafter as practicable. The safety evaluation shall assure that the exposure of offsite personnel to radiation from the primary system coolant activity is within the guidelines of 10 CFR 20.

4. If any reactor coolant leakage exists through a non-isolable fault in a reactor coolant system component (exterior wall of the reactor vessel, piping, valve body, pressurizer or steam generator head), the reactor shall be shut down, and cooldown to the cold shutdown condition shall be initiated within 24 hours of detection. 2

5. The reactor shall not be restarted until the leak is repaired or until the problem is otherwise corrected.

6. When the reactor is critical and above 2 percent power, two
reactor coolant leak detection systems of different operating
principles shall be in operation, with one of the two systems
sensitive to radioactivity. The systems sensitive to radio- 2
activity may be out-of-service for 48 hours provided two
other means are available to detect leakage.

7. Steam generator tube leakage in any one steam generator shall not exceed the limit derived from Figure 15.3.1-1 when averaged over a period of 24 hours.

8. Secondary coolant gas radioactivity shall be monitored continuously 4
by the air ejector gas monitor.

   Secondary coolant gross radioactivity shall be measured weekly. If the air ejector monitor is not operating, the secondary coolant gross radioactivity shall be measured daily to evaluate steam generator leak tightness.

Basis:

Water inventory balances, monitoring equipment, radioactive tracing, boric acid crystalline deposits, and physical inspections can disclose reactor coolant leaks. Any leak of radioactive fluid, whether from the reactor coolant system primary boundary or not can be a serious problem with respect to in-plant radioactivity contamination and cleanup or it could develop into a still more serious problem; and therefore, first indications of such leakage will be followed up as soon as practicable.

Every reasonable effort will be made to reduce reactor coolant leakage to the lowest possible rate and at least below 1 gpm in order to prevent a large leak from masking the presence of a smaller leak. 2
Although some leak rates on the order of 1 gpm may be tolerable from a dose point of view, especially if they are to closed systems, it must be recognized that leaks in the order of drops per minute through any of the walls of the primary system could be indicative of materials failure such as stress corrosion cracking. If depressurization, isolation and/or other safety measures are not taken promptly, these small leaks could develop into much larger leaks, possibly into a gross pipe rupture. Therefore, the nature of the leak, as well as the magnitude of the leakage, must be considered in the safety evaluation. The provision pertaining to a non-isolable fault in a
reactor coolant system component is not intended to cover steam generator 2
tube leakages, valve bonnets or packings, instrument fittings or similar primary system boundaries not indicative of major component exterior wall leakage.

When the source and location of leakage has been identified, the situation can be evaluated to determine if operation can safely continue. This evaluation will be performed by the Manager's Supervisory Staff according to routine established in Section 15.6. Under these conditions, an allowable leakage rate of 10 gpm has been established. The explained leakage rate of 10 gpm is also well within the capacity of one charging pump, and makeup would be available even under the loss of offsite power condition.

If leakage is to the containment, it may be identified by one or more of the following methods:

a. The containment air particulate monitor is sensitive to low leak rates. The rate of leakage to which the instrument is sensitive is 0.013 gpm within 20 minutes, assuming the presence of corrosion product activity.

b. The containment radiogas monitor is less sensitive but can be used as a backup to the air particulate monitor. The sensitivity range of the instrument is approximately 2 gpm to greater than 10 gpm.

c. The humidity detector provides a backup to a. and b. The sensitivity range of the instrumentation is from approximately 2 gpm to 10 gpm.

d. A leakage detection system which determines leakage losses from water and steam systems within the containment collects and measures moisture condensed from the containment atmosphere by cooling coils of the main recirculation units. This system provides a dependable and accurate means of measuring total leakage, including leaks from the cooling coils themselves which are part of the containment boundary. Condensate flows from approximately 1/2 gpm to 10 gpm can be measured by this system.

e. Indication of leakage from the above sources shall be cause to require a containment entry and limited inspection at power of the reactor coolant system. Visual inspection means, i.e., looking for steam, floor wetness or boric acid crystalline formations will be used. Periodic inspections for indications of leakage within the containment will be conducted to enhance early detection of problems and to assure best on-line reliability.

If leakage is to another system, it will be detected by the plant radiation monitors and/or water inventory control.

Steam generator tube leakage limits are based upon offsite dose considerations as limited by 10 CFR 20 in the event of an overpower transient with the presence of collapsed rods and 10 CFR 100 limits in the event of a steam line break or rod ejection accident.

The evaluation of the overpower transient assumed:

a. Five percent of the core iodine inventory is present in the fuel rod gaps.
b. The overpower transient is assumed to fail all flattened rods in the core, and all iodine in the gaps of those rods are 4
immediately released to the coolant.
c. The coolant activity is assumed to leak to the secondary side at a constant rate as given in Figure 15.3.1-1.
d. A retention factor of ten is applied to iodine releases. It is assumed for this analysis that the relief valves remain open for 2 hours following the transient.
e. No activity is released after 2 hours.
f. $X/Q = 3.0 \times 10^{-4}$ sec/m^3.
g. The 2-hour site boundary dose limit is 1.5 Rem thyroid as per 10 CFR 20.

Continuous monitoring of steam generator tube leakage is accomplished by either the Air Ejector Radiation Monitor or the Steam Generator Blowdown Radiation Monitor in combination with periodic surveillance of the primary coolant activity. Backup monitoring can be accomplished by sampling secondary coolant gross activity.

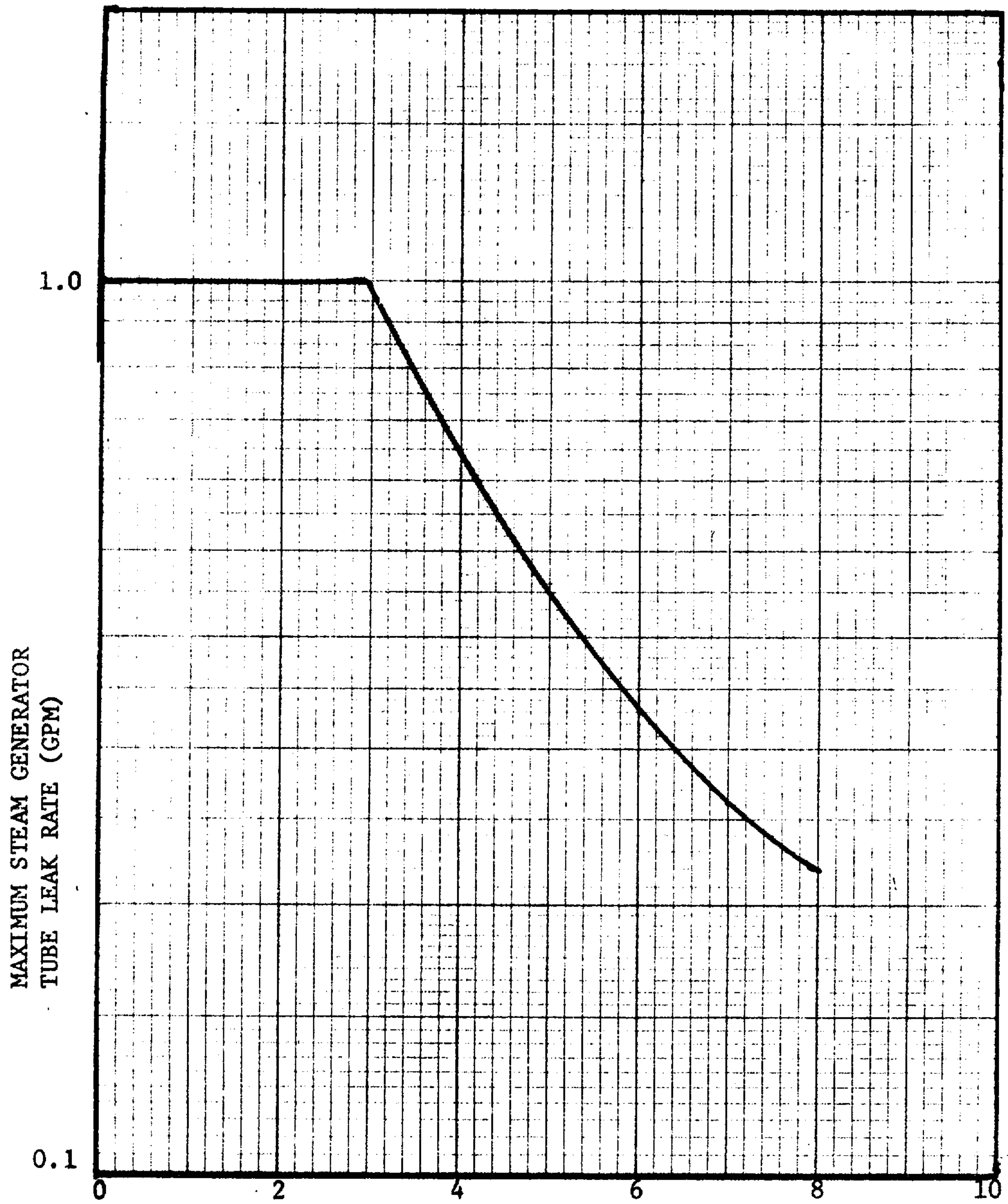
References

FFDSAR Section 6.5, 11.2.3

FIGURE 15.3.1-1

MAXIMUM STEAM GENERATOR LEAK RATE VS. TIME

Based on overpower transient 10 CFR 20 dose limits of 1.5 Rem thyroid



CYCLE 2 TIME (EFPH x 10^{-3})

15.3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

Applicability

Applies to the operation of the control rods and power distribution limits.

Objective:

To ensure (1) core subcriticality after a reactor trip, (2) a limit on potential reactivity insertions from a hypothetical control rod ejection, and (3) an acceptable core power distribution during power operation.

Specification:

A. Control Bank Insertion Limits

1. When the reactor is critical, except for physics tests and control rod exercises, the shutdown control rods shall be fully withdrawn.

2. When the reactor is critical, the control rods shall be inserted no further than the limits shown by the lines on Figure 15.3.10-1 and the shutdown margin with allowance for a stuck rod shall exceed the applicable value shown on Figure 15.3.10-2 under all steady state operating conditions from zero to full power, including effects of axial power distribution. The shutdown margin as used here is defined as the amount by which the reactor core would be subcritical at hot shutdown conditions if all control rods were tripped, assuming that the highest worth control rod remained fully withdrawn and assuming no changes in xenon, boron, or part length rod position. Exceptions to the insertion limit and stuck rod requirements only are permitted for physics tests and control rod exercises.

3. Part length rods shall not be permitted in the core except for low power physics tests and for axial offset calibration tests performed below 75% of rated power.

4. When the reactor is subcritical, except for physics tests, the critical rod position, i.e., the rod position at which criticality would be achieved in the control rods were withdrawn in normal sequence with no other reactivity changes, shall not be lower than the insertion limit for zero power.

B. Power Distribution Limits

1. At all times the hot channel factors defined in the basis must meet the following limits:

   a. $F_Q^N \leq 2.52$ [1 = 0.2 (1-p)] in the indicated flux difference range of +9 to -17 percent.

   $F_{\Delta H}^N \leq 1.58$ [1 + 0.2 (1-P)]

   where P is the fraction of full power at which the core is operating 4

   ($P \leq 1.0$)

   b. If peaking factors exceed the limits of Section B.1.a, the reactor power and high neutron flux trip setpoint shall be reduced by 1 percent for every percent excess over $F_{\Delta H}^N$ or F_Q^N, whichever is limiting. If the peaking factors cannot be corrected within 1 day, the overpower ΔT and overtemperature ΔT setpoints shall be similarly reduced.

   c. The permissible fraction of full power, P, at which the reactor can be operated up to the level of 1518.5 MWt, shall be determined by

$$P = \frac{15.6}{5.7 \times 1.02 \times 1.024 \times 1.007 \times Fq}$$

where $Fq = 2.70 \times L \times \frac{F_{xy}}{1.435} [1 + 2 (T/100 - 0.02)]$

$L = 1$ or 0.96

when surveillance of the axial peaking factor, F_Z^N in Specification B.1.g is in effect; F_{xy} is 1.435, or the value of the unrodded horizontal plane peaking factor appropriate to Fq as determined by a movable incore detector map taken on at least a monthly basis; and T is the percentage operating quadrant tilt limit, having a value of 2 percent if F_{xy} is 1.435 or a value up to 9 percent as selected by the operator if the option to measure F_{xy} is in effect.

d. At rated power, 1518.5 MWt core output, the indicated axial flux difference must be maintained within the range +9 4
percent to -17 percent.

e. For every 4 percent below full power, the permissible positive flux difference range is extended by +1 percent. For every 2.5 percent below full power, the permissible negative flux difference range is extended by - 1 percent.

f. Following initial loading and each subsequent reloading, a power distribution map, using the Movable Detector System, shall be made to confirm that power distribution limits are met, in the full power configuration, before the plant is operated above 75 percent of rating.

g. If the L factor used in Specification B.1.c is 0.96 and the reactor power is ≥ 96 percent of full power, axial surveillance of F_Z^N shall consist of:

(1) Two traverses with the movable incore detectors in appropriate alternate pairs of channels shall be taken every 8 hours, or at a frequency of 0, 10, 30, 60, 120, 180, 240, 360, and 480 minutes following accumulated control rod motion of five steps. From

the traverses, determination that $F_Z \le 1.61/S(Z)$ where S(Z) is the power spike factor given in Figure 15.3.10-3. This allows 4 percent in measurement error.

(2) If a traverse indicates $F_Z > 1.61/S(Z)$, one of the following must be done as soon as practicable, but not exceeding 2 hours after the traverse:

(a) Correct the condition and verify $F_Z \le 1.61/S(Z)$ with two traverses.

(b) If the measured F_Z exceeds 1.61/S(Z), the reactor power shall be reduced by 1 percent for every percent excess over 1.61/S(Z).

2. If the quadrant to average power tilt exceeds a value T% as selected in Specification B.1.c., except for physics testing, then: 4

a. The hot channel factors shall be determined within 2 hours and the power level adjusted to meet the Specification of B.1.a, or

b. If the hot channel factors are not determined within 2 hours, the power and high neutron flux trip setpoint shall be reduced from 100 percent power, 2 percent for each percent of quadrant tilt.

c. If the quadrant to average power tilt exceeds ± 9 percent, except for physics tests, the power level and high neutron flux trip setpoint will be reduced from 100% power, 2% for each percent of quadrant tilt.

3. If after a further period of 24 hours, the power tilt in 2 above is not corrected to less than ± T%, and

a. If design hot channel factors for rated power are not exceeded, an evaluation as to the cause of the discrepancy shall be made and reported as an abnormal occurrence to the Atomic Energy Commission.

b. If the design hot channel factors for rated power are exceeded and the power is greater than 10% - The Atomic Energy Commission shall be notified and the nuclear overpower, overpower ΔT and overtemperature ΔT trips shall be reduced one percent for each percent the hot channel factor exceeds the rated power design values.

c. If the hot channel factors are not determined, the Atomic Energy Commission shall be notified and the overpower
ΔT and overtemperature ΔT trip settings shall be reduced by 4
the equivalent of 2% power for every 1% quadrant to average power tilt.

C. <u>Inoperable Control Rods</u>

1. A control rod shall be considered inoperable if the following occurs:

   a. The rod does not drop upon removal of stationary gripper coil voltage.

b. The rod does not step in properly. It shall be assumed inoperable until it has been tested to verify that it does drop.

c. The rod is shown by the Rod Position Indicator Channel to be misaligned by more than 15 inches. It shall be assumed inoperable until it has been tested to verify that it does step in properly or that it does drop.

2. No more than one inoperable control rod shall be permitted during sustained power operation.

3. When it has been determined that a rod does not drop on removal of stationary gripper coil voltage, the shutdown margin shall be increased by boration as necessary to compensate for the withdrawn worth of the inoperable rod. If sustained power operation is anticipated, the rod insertion limit shall be adjusted to reflect the worth of the inoperable rod.

D. Misaligned or Dropped Control Rod

1. If the Rod Position Indicator Channel is functional and the associated part length or full length control rod is more than 15 inches out of alignment with its bank and cannot be realigned, then unless the hot channel factors are shown to be within design limits as specified in Section 15.3.10.B-1 within 8 hours, power shall be reduced so as not to exceed 75% of rated power.

2. To increase power above 75% with a part-length or full length control rod more than 15 inches out of alignment with its bank an analysis shall first be made to determine the hot channel factors and the resulting allowable power level based on Section 15.3.10.B.

3. If it be determined that the apparent misalignment or dropped rod indication was caused by Rod Position Indicator Channel failure, sustained power operation can be continued if the following conditions are met:

   a. For operation between 10% power and rated power, the position of the rod(s) with the failed Rod Position Indicator Channel(s) will be checked indirectly by core instrumentation (excore detectors, and/or thermocouples, and/or moveable incore detectors) every shift or after associated bank motion exceeding 24 steps, whichever comes sooner.

   b. For operation below 10% of rated power, no special monitoring is required.

E. Rod Drop Times

1. At operating temperature and full flow, the drop time of each control rod shall be no greater than 1.8 seconds from the loss of stationary gripper coil voltage to dashpot entry.

Basis:

The reactivity control concept is that reactivity changes accompanying changes in reactor power are compensated by control rod motion. Reactivity changes associated with xenon, samarium, fuel depletion, and large changes in reactor coolant temperature (operating temperature to cold shutdown) are compensated by changes in the soluble boron concentration. During power operation, the shutdown groups are fully withdrawn and control of reactor power is by the control groups. A reactor trip occuring during power operation will put the reactor into the hot shutdown condition. The control rod insertion limits provide for achieving hot shutdown by reactor trip at any time and assume the highest worth control rod remains fully withdrawn. The rods are withdrawn in the sequence of A, B, C, D with overlap between banks and a 10% margin in reactivity worth of the control rods to assure meeting the assumptions used in the accident analysis. In addition, they provide a limit on the maximum inserted rod worth in the unlikely event of a hypothetical rod ejection, and provide for acceptable nuclear peaking factors. The solid lines shown on Figure 15.3.10-1 meet the shutdown requirement. The maximum shutdown margin requirement occurs at end of core life and is based on the value used in analysis of the hypothetical steam break accident. Early in core life, less shutdown margin is required, and Figure 15.3.10-2 shows the shutdown margin equivalent to 2.77% reactivity at end-of-life with respect to an uncontrolled cooldown. All other accident analyses are based on 1% reactivity shutdown margin.

The specified control rod insertion limits have been revised for cycle 2, in order to meet the design basis criteria on (1) potential ejected control rod worth and peaking factor, (2) radial power peaking factors, $F^N_{\Delta H}$, and (3) required shutdown margin. 4

The overlap between successive control banks is provided to compensate for the low differential rod worth near the top and bottom of the core. Part length rod insertion has been eliminated for cycle 2 to eliminate certain adverse power changes and to preclude rapid local power changes caused by part length rod travel through the 4
core.

The various control rod banks (shutdown rods, control banks A, B, C, D, and part length rods) are each to be moved as a bank, that is, with all rods in the bank within one step (5/8-inch) of the bank position. Direct information on rod position indication is provided by two methods: a digital count of actuating pulses which shows the demand position of the banks and a linear position indicator (LVDT) which indicates the actual rod position. The Rod Position Indicator channel has a demonstrated accuracy of 5% of span (7.2 inches). Therefore, a 15-inch indicated misalignment of a rod from its bank is necessarily a true misalignment. Misalignment of 15 inches cannot cause design hot channel factors to be exceeded, and complete rod misalignment (part-length or full-length control rod 12 feet out of alignment with its bank) does not result in exceeding core limits in steady-state operation at rated power. If the misalignment condition cannot be readily corrected, the specified reduction in power to 75% will insure that design margins to core limits will be maintained under both steady-state and anticipated transient conditions. The 8-hour permissible limit on rod misalignment at rated power is short with respect to the probability of an independent accident. The failure of an LVDT in itself does not reduce the shutdown capability of the rods, but it does reduce the operator's capability for determining the position of that rod by direct means. The operator has available to him the core detector recordings, incore thermocouple readings and periodic incore flux traces for indirectly determining rod position and flux tilts should the rod with the inoperable LVDT become malpositioned. The excore and incore instrumentation will not necessarily recognize a misalignment of 15 inches be because the concommitant increase in power density will normally be less than 1% for a 15-inch misalignment. The excore and incore instrumentation will, however, detect any rod misalignment which is sufficient to cause a significant increase in hot channel factors and/or any significant loss in shutdown capability. The increased surveillance of the core if one or more Rod Position Indicator Channels is out of service serves to guard against any significant loss in shutdown margin or margin to core thermal limits. The history of malpositioned rods indicates that in nearly all the cases when the rods have been malpositioned, the malpositioning occurred when the bank was moving. The checking of the rod position after bank motion exceeding 24 steps will verify that the rod with the inoperable LVDT is moving properly with its bank and according to the bank step counter. Malpositioning of a rod in a bank which is not moving is very rare, and, if it does occur, it is usually gross slippage or complete rod dropping which will be seen by external detectors. Should it go undetected, the checking of the rod position every shift is short with respect to the probability of another independent undetected situation which would further reduce the shutdown capability of the rods. Any combination of misaligned rods below 10% rated power will not exceed the design limits. For this reason, the position of the rods with inoperable LVDT's need not be checked below 10% power; plus, the incore instrumentation is not effective for determining rod position until the power level is above approximately 5%.

An inoperable rod imposes additional demands on the operators, the permissible number of inoperable control rods is limited to one in order to limit the magnitude of the operating burden. From operating experience to date, a control rod which steps "in" properly will drop when a trip signal occurs because the only force acting to drive the rod in is gravity. When it has been determined that a rod does not drop, extra margin is gained by boration or by adjusting the insertion limit to account for the worth of the inoperable control rod.

Two criteria have been chosen as a design basis for fuel performance related to fission gas release, pellet temperature and cladding mechanical properties. First the peak value of linear power density must not exceed 18.1 kW/ft. Second, the minimum DNBR in the core must not be less than 1.30 in normal operation or in short term transients.

In addition to the above, the initial steady state conditions for the peak linear power for a loss-of-coolant accident must not exceed the values assumed in the accident evaluation. This limit is required in order for the maximum clad temperature to remain below that established by the Interim Policy Statement for LOCA, and below those limits prescribed in the AEC Regulatory staff's "Technical Report on Densification in Light Water Reactor Fuels."

To aid in specifying the limits on power distribution the following hot channel factors are defined.

F_Q, Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod divided by the average fuel 4
rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods.

F_Q^N, Nuclear Heat Flux Hot Channel Factor, is defined as the maximum local fuel rod linear power density divided by the average fuel rod linear power density, assuming nominal fuel pellet and rod dimensions.

F_Q^E, Engineering Heat Flux Hot Channel Factor, is defined as the ratio between F_Q, and F_Q^N and is the allowance on heat flux required for manufacturing tolerances.

$F_{\Delta H}^N$, Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod on which minimum DNBR occurs to the average rod power.

It should be noted that $F^N_{\Delta H}$ is based on an integral and is used as such in the DNB calculations. Local heat fluxes are obtained by using 4
hot channel and adjacent channel explicit power shapes which take into account variations in horizontal (x-y) power shapes throughout the core. Thus the horizontal power shape at the point of maximum heat flux is not necessarily directly related to $F^N_{\Delta H}$.

It has been determined by analysis that the design limits on peak local power density on minimum DNBR at full power and LOCA are met, provided:

$$F_Q^N \leq 2.52 \text{ and } F_{\Delta H}^N \leq 1.58$$

These quantities are measurable although there is not normally a requirement to do so. Instead it has been determined that, provided certain conditions are observed, the above hot channel factor limits will be met at full power; these conditions are as follows.

1. Control rods in a single bank move together with no individual rod insertion differing by more than 15 inches from the bank demand position.

2. Control rod banks are sequenced with overlapping banks as shown in Figure 15.3.10-1. 4

3. The control bank insertion limits are not violated.

4. Axial power distribution guide lines, which are given in terms of flux difference control are observed. Flux difference refers to the difference in signals between the top and bottom halves of two-section excore neutron detectors. The flux difference is a measure of the axial offset which is defined as the difference in power between the top and bottom halves of the core. Calculation of core peaking factors under a variety of operating conditions have been correlated with axial offset. The correlation shows that an F_Q^N of 2.52 and allowed DNB shapes, including the effects of fuel densification, are not exceeded if the axial offset (flux difference) is maintained between -20 and +12%. The specified limits of -17 and +9% allow for a 3% error in

the axial offset. In order to take credit for operation at the bounding value of the correlation in the permitted range of the axial offset, surveillance of the axial peaking factor, F_Z, is specified. Otherwise the specification leads to a 4% penalty in power.

For operation at a fraction, P, of full power the design limits are met, provided, 4

$F_Q^N \leq 2.52\ [1 + 0.2(1-P)]$ in the indicated flux difference range of +9 to -17

and $F_{\Delta H}^N \leq 1.58\ [1 + 0.2\ (1-P)]$

The permitted relaxation allows radial power shape changes with rod insertion to the insertion limits. It has been determined that provided the above conditions 1 through 4 are observed, these hot channel factors limits are met.

For normal operation and anticipated transients the core is protected from exceeding 18.1 kw/ft locally, and from going below a minimum DNBR of 1.30, by automatic protection on power, flux difference, pressure and temperature. Only conditions 1 through 3, above, are mandatory since the flux difference is an explicit input to the protection system.

Measurements of the hot channel factors are required as part of startup physics tests and whenever abnormal power distribution conditons require a reduction of core power to a level based on measured hot channel factors.

In the specified limit of F_Q^N there is a 5% allowance for uncertainties[1]
which means that normal operation of the core within the defined conditions 4
and procedures is expected to result in $F_Q^N \leq$ 2.52/1.05 even on a worst

case basis. When a measurement is taken experimental error must be allowed for and 5% is the appropriate allowance for a full core map taken with the moveable incore detector flux mapping system.

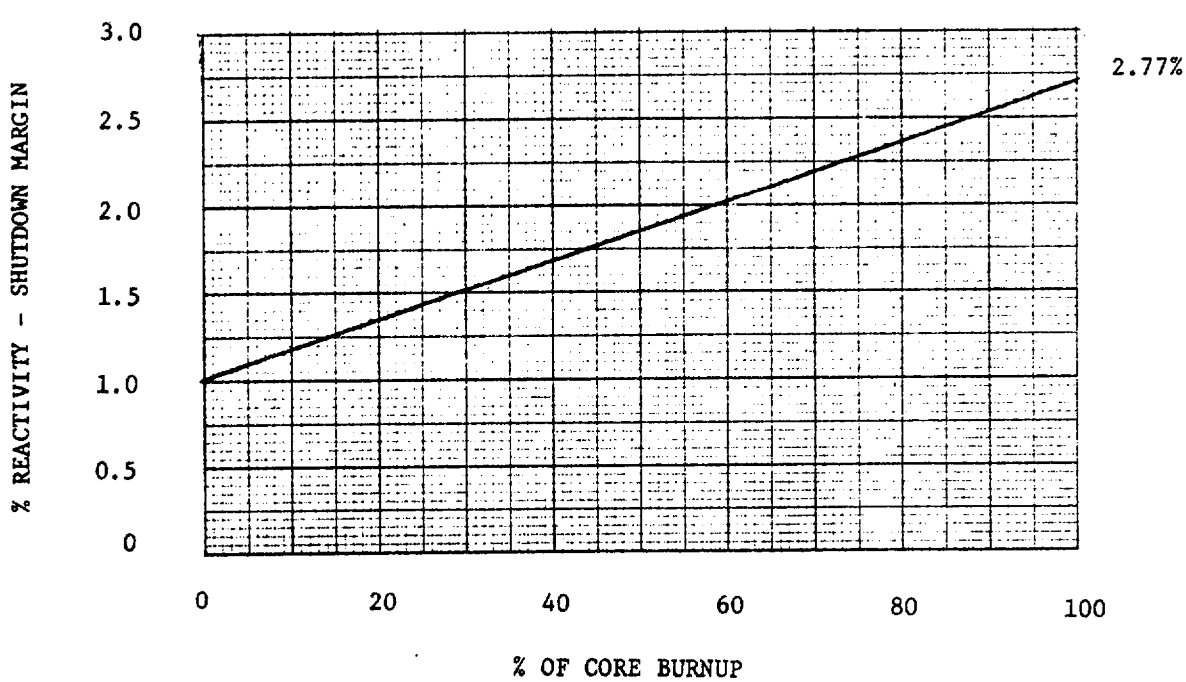
The measured value of F_Q^N must be additionally corrected by including a penalty as shown on Figure 15.3.10-3 (at the appropriate core location) to account for fuel densification effects before comparison with the limiting value above.

In the specified limit of $F_{\Delta H}^N$ there is a 10% allowance for uncertainties[1]
which means that normal operation of the core is expected to result in
$F_{\Delta H}^N \leq 1.58/1.10$. The logic behind the larger uncertainty in this case 4
is that (a) abnormal perturbations in the radial power shape (e.g., rod
misalignment) affect $F_{\Delta H}^N$, in most cases without necessarily affecting
F_Q^N, through movement of part length rods, and can limit it to the desired
value, he had no direct control over $F_{\Delta H}^N$ and (c) an error in the predictions
for radial power shape, which may be detected during startup physics
tests can be compensated for in F_Q^N by tighter axial control, but com-
pensation for $F_{\Delta H}^N$ is less readily available. Five percent is the
appropriate allowance for a full core map taken with the moveable
incore detector flux mapping system.

At the option of the operator, credit may be taken for measured decreases in the unrodded horizontal plane peaking factor, F_{xy}. This credit may take the form of a reduction in F_Q or expansion of permissible quadrant tilt limits over the 2% value, up to a value of 9%, at which point specified power reductions are prudent. Monthly surveillance of F_{xy} bounds the quantity because it decreases with burnup (WCAP-7912L)

A 2% quadrant tilt allows that a 5% tilt might actually be present in the core because of insensitivity of the excore detectors for disturbances near the core center such as misaligned inner control rods and an error allowance. No increase in F_Q occurs with tilts up to 5% because misaligned control rods producing such tilts do not extend to the unrodded plane, where the maximum F_Q occurs. 4

The Point Beach Unit 1 densification report justifies in Section 5.4 the factors in the denominator of the equation defining the maximum permissible fraction of full power. Credit for the final factor of 1.02 in that section is not given in the specification because of the assumptions in the loss of flow accident in Section 6.6 of the above report.



% REACTIVITY - SHUTDOWN MARGIN
3.0
2.5
2.0
1.5
1.0
0.5
0
2.77%
0
20
40
60
80
100
% OF CORE BURNUP

FIGURE 15.3.10-3
POWER SPIKE FACTOR VS. ELEVATION

POINT BEACH 1 - CYCLE 2
ASSUMES CLAD
FLATTENING