Westinghouse Electric Corporation    Power Systems

PWR Systems Division

Box 355
Pittsburgh Pennsylvania 15230

October 24, 1980

NS-TMA-2323

Mr. James R. Miller, Chief
Special Projects Branch
Division of Project Management
U. S. Nuclear Regulatory Commission
Phillips Building
7920 Norfolk Avenue
Bethesda, Maryland 20014

Subject: Responses to "Request for Additional Information on WCAP-9500 Section 4.3 and Chapter 16", NRC letter from J. R. Miller to T. M. Anderson, August 2, 1980

Dear Mr. Miller:

Enclosed are:

1. Forty (40) copies of the proprietary responses to the NRC Request for additional information on WCAP-9500, Section 4.3 and Chapter 16.

2. Thirty-five (35) copies of the non-proprietary responses to the NRC request for additional information on WCAP-9500, Section 4.3 and Chapter 16.

Also enclosed are:

1. One (1) copy of Application for Withholding (Non-Proprietary).

2. One (1) copy of original Affidavit (Non-Proprietary).

This submittal contains proprietary information of Westinghouse Electric Corporation. In conformance with the requirements of 10CFR2.790, as amended, of the Commission's regulations, we are enclosing with this submittal an application for withholding from public disclosure and an affidavit. The affidavit sets forth the basis on which the information may be withheld from public disclosure by the Commission.

NEED 2 RIDS

T007 S 1/40

T008 S 1/35



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Correspondence with respect to the affidavit or application for withholding should reference AW-80-64 and should be addressed to R. A. Wiesemann, Manager of Regulatory and Legislative Affairs, Westinghouse Electric Corporation, P. O. Box 355, Pittsburgh, Pennsylvania 15230.

Very truly yours,

T. M. Anderson, Manager
Nuclear Safety Department

CJR/kk
Enclosures

Westinghouse Electric Corporation
Water Reactor Divisions
Nuclear Technology Division
Box 355
Pittsburgh Pennsylvania 15230

October 24, 1980
AW-80-64

Mr. James R. Miller, Chief
Special Projects Branch
Division of Project Management
U. S. Nuclear Regulatory Commission
Phillips Building
7920 Norfolk Avenue
Bethesda, Maryland 20014

APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

SUBJECT: Proprietary Responses to "Request for Additional Information on WCAP-9500 Section 4.3 and Ch. 16," NRC letter from J. R. Miller to T. M. Anderson, August 2, 1980

REF: Westinghouse Letter No. NS-TMA-2323, Anderson to Miller, dated October 24, 1980

Dear Mr. Miller:

The proprietary material transmitted by the referenced letter is of the same technical type as material previously submitted concerning the Westinghouse optimized fuel assembly program (Reference: NS-TMA-2057, dated March 30, 1979). Further, the affidavits submitted to justify the material previously submitted, AW-78-23 and AW-78-61, are equally applicable to this material.

Accordingly, withholding the subject information from public disclosure is requested in accordance with the previously submitted affidavit and application for withholding, AW-78-23, dated March 21, 1978, a copy of which is attached.

Correspondence with respect to this application for withholding or the accompanying affidavit should reference AW-80-64, and should be addressed to the undersigned.

Very truly yours,

Robert A. Wiesemann

Robert A. Wiesemann, Manager
Regulatory & Legislative Affairs

wpc
Attachment

cc: E. C. Shomaker, Esq.
Office of the Executive Legal Director, NRC

AW-78-23

POOR ORIGINAL

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared Robert A. Wiesemann, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Westinghouse Electric Corporation ("Westinghouse") and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

Robert A. Wiesemann

Robert A. Wiesemann, Manager
Licensing Programs

Sworn to and subscribed
before me this 20 day
of March 1978.

Notary Public

(1) I am Manager, Licensing Programs, in the Pressurized Water Reactor Systems Division, of Westinghouse Electric Corporation and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing or rulemaking proceedings, and am authorized to apply for its withholding on behalf of the Westinghouse Water Reactor Divisions.

(2) I am making this affidavit in conformance with the provisions of 10 CFR Section 2.790 of the Commission's regulations and in conjunction with the Westinghouse application for withholding accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse Nuclear Energy Systems in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.790 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitutes Westinghouse policy and provides the rational basis required.

## Criteria and Standards Utilized

In determining whether information in a document or report is proprietary, the following criteria and standards are utilized by Westinghouse. Information is proprietary if any one of the following are met:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

(g) It is not the property of Westinghouse, but must be treated as proprietary by Westinghouse according to agreements with the owner.

(iii) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.790, it is to be received in confidence by the Commission.

(iv) The information is not available in public sources to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld in this submittal are the copies of slides utilized by Westinghouse in its presentation to the NRC at the March 21, 1978 meeting concerning the Westinghouse optimized fuel assembly. The letter and the copies of slides are being submitted in preliminary form to the Commission for review and comment on the Westinghouse optimized fuel assembly in advance of a formal submittal for NRC approval.

Public disclosure of this information is likely to cause substantial harm to the competitive position of Westinghouse as it would reveal the description of the approved design, the comparison of the improved design with the standard design, the nature of the tests conducted, the test conditions, the test results and the conclusions of the testing program, all of which is recognized by the Staff to be of competitive value and because of the large amount of effort and money expended by Westinghouse over a period of several years in carrying out this particular development program. Further, it would enable competitors to use the information for commercial purposes and also to meet NRC requirements for licensing documentation, each without purchasing the right from Westinghouse to use the information.

Information regarding its development programs is valuable to Westinghouse because:

(a) Information resulting from its development programs gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information which is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

Being an innovative concept, this information might not be discovered by the competitors of Westinghouse independently. To duplicate this information, competitors would first have to be similarly inspired and would then have to expend an effort similar to that of Westinghouse to develop the design.

Further the deponent sayeth not.

Question:

1. Provide a complete description of how power distribution limits are met in normal opertion of the power plant. Available references do not tell the complete story. For example, under what conditions is use of the APDMS required? Or, when is an 18 case or subset analysis of CAOC required? Or, how is the K(z) curve generated when $F_Q$ is not 2.32?

Response:

To satisfy the Final Acceptance Criteria (10 CFR 50.46) for the Loss of Coolant Accident, the total core peaking factor ($F_Q$ times relative power) is evaluated as a function of core height and compared to the Technical Specification Limit. The models and synthesis procedures used to perform this evaluation are described in Reference [1].

All of the nuclear effects which influence axial power distributions throughout the fuel cycle are included in the evaluation of the total peaking factor. Various modes of load follow and base load operation are considered. This evaluation is based on normal plant operation in compliance with the Technical Specifications.

For cores that operate within the limits of Constant Axial Offset Control (CAOC), the evaluation is initiated by determining whether the core operates within the following constraints:

1. The Technical Specification limit on the maximum height dependent $F_Q$ is greater than or equal to a value of 2.32, and

2. The CAOC flux difference ($\Delta I$) bandwidth is less than or equal to $\pm 5$ percent $\Delta I$.

+[ (a,c)

]+ (a,c)

If a core that operates without part-length rods and within the limits of CAOC cannot satisfy either of the above conditions, then a set of plant maneuvers, somewhat different than those presented in Reference [1], are studied to determine the limiting height dependent total core peaking factor during the fuel cycle. Table 1 lists the maximum number of cases [ ]+ which would be evaluated under the above circumstances. These cases are discussed in Reference [2]. If the plant has acquired the Westinghouse Improved Load Follow Package (ILFP) a modified set of cases has been selected for analysis. These cases are discussed in Reference [3]. (a,c)

+[ (a,c)

]+

+[

(a,c)

]* (a,c)

Due to a change in the LOCA analysis, the Peak Clad Temperature (PCT) may exceed the 2200°F limit if the $F_Q$ limit is not set at a lower value. To ensure that the PCT is maintained at an acceptable level, a plant may be restricted to operating with an $F_Q$ value below the standard value of 2.32, thus requiring the generation of a new $F_Q$ envelope.

The revised $F_Q$ envelope is drawn similarly to the standard 2.32 envelope, with the third line segment in common. The first line segment is drawn as in $F_Q = 2.32$ but using the new $F_Q$ limit to define the endpoints as $(0, F_Q)$ and $(6, F_Q)$. The small break analysis, which defines the third line segment, is not usually redone. The previous $F_Q$ envelope third line segment is used and the applicable portion is between the second and third line segment intercept and the 12 ft. endpoint. If the small break analysis is redone, the new third line segment information will be provided and the $F_Q$ envelope will be drawn as described.

The second line segment originates at $(6, F_Q)$, has a defined value at $Z = 10.8$ ft., and terminates at the intersection with the third line segment. To determine the $F_Q$ value at $Z = 10.8$ ft., the new $F_Q$ limit must be normalized to the standard $F_Q$ value. This point is defined as $(10.8, 2.18/2.32\ F_Q)$ and is to be used, for this purpose, to determine the slope of the second line segment. The intercept point is calculated to completely define the bounds of the envelope. The intercept is determined by solving two simultaneous equations, which are defined as:

$$\frac{a-y}{12-x} = m_3 \qquad \text{and} \qquad \frac{y-b}{x-10.8} = m_2$$

where:

- a = y coordinate of third line segment endpoint
- b = $\frac{2.18}{2.32} F_Q$
- x = x coordinate of intercept point
- y = y coordinate of intercept point
- $m_2$ = slope of second line segment
- $m_3$ = slope of third line segment

The K(z) function is defined by normalizing the $F_Q$ values at any core height to the first line segment value. This function appears in the Technical Specifications and is used to determine the allowable $F_Q$ at any operating condition.

In most instances, the use of the [ ]+ analysis results in a calculated $F_Q$ * P that is below the allowable limit. However, in some plants with unusually low allowable $F_Q$ limits, a violation of the limit may occur. In these plants the use of the Axial Power Distribution Monitoring System (APDMS) is required. The APDMS is a surveillance tool that utilizes the standard incore moveable detectors to verify compliance with technical specification limits on the total peaking factor, $F_Q$. The APDMS is required at or above the power level where potential violations of the $F_Q$ envelope may occur. This turn on power (in fraction of RTP) is defined as the ratio of the maximum allowable $F_Q$ * P divided by the maximum calculated $F_Q$ * P. (a,c)

TABLE 1

Summary of [ ]

+(a,c)

| Case* |
|---|
| 1 |
| 1 |
| 4 |
| 4 |
| 7 |
| 7 |

*Case described on Table 4-1 of Reference [1]

## REFERENCES

1. T. Morita, et. al. "Power Distribution Control and Load Following Procedures," WCAP-8385 (P) and WCAP-8403 (non-P), September, 1974.

2. "$F_Q$ Envelope Calculations," C. E. Eicheldinger letter NS-CE-687 to D. B. Vassallo (NRC) dated June 27, 1975.

3. T. M. Anderson letter NS-TMA-2198 to K. Kniel (NRC) dated January 31, 1980.

4. "Justification of Peaking Factor Subcase Analysis," C. E. Eicheldinger letter NS-CE-1749 to J. F. Stolz (NRC) dated April 6, 1978.

5. K. A. Jones, et. al., "Axial Power Distribution Monitoring System," WCAP-8589 (P), August, 1975.

Question:

2. Provide additional information justifying the coefficient change to 0.3 of the part power $F_{\Delta H}^{N}$ allowance shown on (blue) p. 4.3-26 and p. 4.4-35.

Response:

Increasing allowable $F_{\Delta H}^{N}$ with decreasing power is permitted by all previously approved Westinghouse designs. The increase is permitted by the DNB protection setpoints and allows radial power shape changes with rod insertion to the insertion limit, as described in Section 4.3.2.2.6 of the Reference Core Report for the 17x17 Optimized Fuel Assembly (WCAP-9500). The design limit of the nuclear enthalpy rise factor is given on p. 4.3-26 (blue) as:

$$\text{limit } F_{\Delta H}^{N} = 1.55 [1 + 0.3 (1-P)]$$

where P is the fraction of rated full power.

The maximum calculated value of the operating nuclear enthalpy rise factor as a function of power level, including uncertainty allowance, does not exceed the design limit at any power level. This is demonstrated in Figure 1 which compares the calculated enthalpy rise factor against the design limit, as a function of power level. The calculated enthalpy rise factors shown in Figure 1 are conservative values that assume insertion of control rods to the control rod insertion limits given in the Technical Specifications (Chapter 16 of WCAP-9500) and assume the most limiting axial power distributions allowable within the Constant Axial Offset Control (COAC) operation limits defined in the Technical Specifications.

Figure 1

Conservative Calculation of Enthalpy Rise Factor
With Power Level and Technical Specifications Limit

+ a,c

Question:

3. On p. 4.3-27 reference is made to "administrative controls and alarms . . . provide for returning the core to a safe condition." Provide a description of all of these administrative controls and in particular more detail on why each alarm is provided and how each alarm is derived. The latter is intended to be at a level which will permit relative evaluation of the quality of the alarms, not an instrumentation and control review.

Question:

4. There are many references to Chapter 7 and 16 in Section 4.3 of WCAP-9500. Please eliminate these references, and provide the appropriate description in Section 4.3, or at least make the references very specific. This more clearly applies to Chapter 16 than 7. It is recognized that some of the references to Chapter 7 may be general.

Response:

The attached text has been amended in response to Questions 3 and 4.

The maximum reactivity worth of control rods and the maximum rates of reactivity insertion employing control rods are limited so as to preclude rupture of the coolant pressure boundary or disruption of the core internals to a degree which would impair core cooling capacity due to a rod withdrawal or ejection accident (see Chapter 15).

Following any Condition IV event (rod ejection, steamline break, etc.) the reactor can be brought to the shutdown condition and the core will maintain acceptable heat transfer geometry. This satisfies GDC-28.

Discussion

Reactivity addition associated with an accidental withdrawal of a control bank (or banks) is limited by the maximum rod speed (or travel rate) and by the worth of the bank(s). The maximum control rod speed is 45 inches per minute and the maximum rate of reactivity change considering two control banks moving is less than 75 pcm/sec. During normal operation at power and with normal control rod overlap, the maximum reactivity change rate is less than 45 pcm/sec.

The reactivity change rates are conservatively calculated assuming unfavorable axial power and xenon distributions. The peak xenon burnout rate is 25 pcm/min, significantly lower than the maximum reactivity addition rate of 45 pcm/sec for normal operation and 75 pcm/sec for accidental withdrawal of two banks.

4.3.1.5 Shutdown Margins

Basis

Minimum shutdown margin as required in specifications 3/4.1.1.1 and 3/4.1.1.2 of the technical specifications is required at any power operating condition, in the hot standby and hot shutdown conditions and in the cold shutdown condition.

In all analysis involving reactor trip, the single, highest worth rod cluster control assembly is postulated to remain untripped in its full-out position (stuck rod criterion). This satisfies GDC-26.

The boron concentration required to meet the refueling shutdown criteria is noted in specification 3/4.9.1 of the Technical Specifications. Verification that this shutdown criteria is met, including uncertainties, is achieved using standard Westinghouse design methods such as LEOPARD (Reference [19]), TURTLE (Reference [10]) a diffusion theory code, and PALADON (Reference [38]) a nodal analysis code. The subcriticality of the core is continuously monitored as described in specification 3/4.9.1 of the Technical Specifications.

#### 4.3.1.6 Stability

Basis

The core will be inherently stable to power oscillations at the fundamental mode. This satisfies GDC-12. Spatial power oscillations within the core with a constant core power output, should they occur, can be reliably and readily detected and suppressed.

Discussion

Oscillations of the total power output of the core, from whatever cause, are readily detected by the N-16 power detectors, the loop temperature sensors and by the nuclear instrumentation. The core is protected by these systems and a reactor trip would occur (primarily from the P(z) portion of the N-16 high kw/ft reactor trip), if power increased unacceptably, preserving the design margins to fuel design limits. The stability of the turbine/steam generator/core systems and the reactor control system is such that total core power oscillations are not normally possible. The redundancy of the protection circuits ensures an extremely low probability of exceeding design power levels.

The core is designed so that diametral and azimuthal oscillations due to spatial xenon effects are self-damping and no operator action or control action is required to suppress them. The stability to diametral oscillations is so great that this excitation is highly improbable. Convergent azimuthal oscillations can be excited by prohibited motion of individual control rods. Such oscillations are readily observable and alarmed (on kw/ft above limit), using the multisection excore ion detectors. Indications are also continuously available from incore thermocouples and loop

The boron concentration required to meet the refueling shutdown criteria is noted in specification 3/4.9.1 of the Technical Specifications. Verification that this shutdown criteria is met, including uncertainties, is achieved using standard Westinghouse design methods such as LEOPARD (Reference 19), TURTLE (Reference 10) a diffusion theory code, and PALADON (Reference 38) a nodal analysis code. The subcriticality of the core is continuously monitored as described in specification 3/4.9.1 of the Technical Specifications.

#### 4.3.1.6 Stability

Basis

The core will be inherently stable to power oscillations at the fundamental mode. This satisfies GDC-12. Spatial power oscillations within the core with a constant core power output, should they occur, can be reliably and readily detected and suppressed.

Discussion

Oscillations of the total power output of the core, from whatever cause, are readily detected by the loop temperature sensors and by the nuclear instrumentation. The core is protected by these systems and a reactor trip would occur (primarily from the ΔI portion of the Overtemperature ΔT reactor trip), if power increased unacceptably, preserving the design margins to fuel design limits. The stability of the turbine/ steam generator/core systems and the reactor control system is such that total core power oscillations are not normally possible. The redundancy of the protection circuits ensures an extremely low probability of exceeding design power levels.

The core is designed so that diametral and azimuthal oscillations due to spatial xenon effects are self-damping and no operator action or control action is required to suppress them. The stability to diametral oscillations is so great that this excitation is highly improbable. Convergent azimuthal oscillations can be excited by prohibited motion of individual control rods. Such oscillations are readily observable and alarmed (by $\Delta I$ outside of the CAOC target band), using the excore long ion chambers. Indications are also continuously available from incore thermocouples and loop temperature thimble. Figure 4.2-1 shows a cross sectional view of a 17 x 17 fuel assembly and the related rod cluster control locations. Further details of the fuel assembly are given in Section 4.2.

The fuel rods within a given assembly have the same uranium enrichment in both the radial and axial planes. Fuel assemblies of three different enrichments are used in the initial core loading to establish a favorable radial power distribution. Figure 4.3-1 shows the fuel loading pattern to be used in the first core. Two regions consisting of the two lower enrichments are interspersed so as to form a checkerboard pattern in the central portion of the core. The third region is arranged around the periphery of the core and contains the highest enrichment. The enrichments for the first core are shown in Table 4.3-1.

The reference reloading pattern is typically similar to Figure 4.3-1 with depleted fuel interspersed checkerboard style in the center and new fuel on the periphery. The core will normally operate approximately one year between refuelings, accumulating approximately 12,000 MWD/MTU per year. The exact reloading pattern, initial and final positions of assemblies, number of fresh assemblies and their placement are dependent on the energy requirement for the next cycle and burnup and power histories of the previous cycles.

The core average enrichment is determined by the amount of fissionable material required to provide the desired core lifetime and energy requirements, namely a region average discharge burnup of 36,000 MWD/MTU. The physics of the burnout process is such that operation of the reactor depletes the amount of fuel available due to the absorption of neutrons by the U-235 atoms and their subsequent fission. The rate of U-235 depletion is directly proportional to the power level at which the reactor is operated. In addition, the fission process results in the formation of fission products, some of which readily absorb neutrons. These effects, depletion and the buildup of fission products, are partially offset by the buildup of plutonium shown in Figure 4.3-2 for the 17 x 17 fuel assembly, which occurs due to the non-fission absorption of neutrons in U-238. Therefore, at the beginning of any

The means for maintaining power distributions within the required hot channel factor limits are described in the Surveillance and Action requirements of specifications 3/4.2.1 and 3/4.2.2 (3/4.2.1, 3/4.2.2, and 3/4.2.3 Blue) of the Technical Specifications. A complete discussion of power distribution control in Westinghouse PWRs is included in Reference [6]. Detailed background information on the design constraints on local power density in a Westinghouse PWR, on the defined operating procedures and on the measures taken to preclude exceeding design limits is presented in the Westinghouse Topical Report on power distribution control and load following procedures (Reference [7]). The following paragraphs summarize these reports and describe the calculations used to establish the upper bound on peaking factors.

The calculations used to establish the upper bound on peaking factors, $F_Q$ and $F^N_{\Delta H}$, include all of the nuclear effects which influence the radial and/or axial power distributions throughout core life for various modes of operation including load follow, reduced power operation, and axial xenon transients.

Radial power distributions are calculated for the full power condition and fuel and moderator temperature feedback effects are included for the average enthalpy plane of the reactor. The steady state nuclear design calculations are done for normal flow with the same mass flow in each channel and flow redistribution is calculated explicitly where it is important in the DNB analysis of accidents. The effect of xenon on radial power distribution is small (compare Figures 4.3-6 and 4.3-7) but is included as part of the normal design process. Radial power distributions are relatively fixed and easily bounded with upper limits.

The core average axial profile, however, can experience significant changes which can occur rapidly as a result of rod motion and load changes and more slowly due to xenon distribution. For the study of points of closest approach to axial power distribution limits, several thousand cases are examined. Since the properties of the nuclear design dictate what axial shapes can occur, boundaries on the limits of interest can be set in terms of the parameters which are readily observed on the plant. Specifically, the nuclear design parameters which are significant to the axial power distribution analysis are:

Using these procedures, the calculated points are synthesized from axial calculations combined with radial factors appropriate for rodded and unrodded planes in the first cycle. In these calculations the effects on the unrodded radial peak of xenon redistribution that occurs following the withdrawal of a control bank (or banks) from a rodded region is obtained from two-dimensional XY calculations. A 1.03 factor to be applied on the unrodded radial peak was obtained from calculations in which xenon distribution was preconditioned by the presence of control rods and then allowed to redistribute for several hours. A detailed discussion of this effect may be found in Reference [7]. The calculated values have been increased by a factor of 1.05 for conservatism and a factor of 1.03 for the engineering factor $F_Q^E$.

The results demonstrate that the design basis limits of $F_Q$ (Z) times relative-power shown in Figure 4.3-21 provides a conservative upper bound for any cycle of operation. This method of analysis, however, is no longer necessary since compliance with the design envelope will be demonstrated by the peak linear power density surveillance system as described in Sections 7, 15 and specification 3/4.2.1 of 16.

Finally, as previously discussed, normal operation is based on manual or automatic operating procedures for base load and load follow operation. These procedures require computer based surveillance supplemented by the normal periodic full core map requirement and a computer-based alarm (on axial flux difference deviation or high kw/ft) for violations of the design limit envelope.

The reactor KW/ft protection system setpoints are adjusted to prevent the peak linear power density from exceeding 18 KW/ft for Condition II events e.g., rod control equipment malfunction, operator errors of commission and operator errors of omission. The direct KW/ft and DNB protection eliminates the historical need for the detailed overpower analyses described in Reference [7] to demonstrate compliance with DNB and peak linear power density limits based on a correlation between hot channel factors and axial offset.

The key nuclear inputs to the protection system are the methods for generating $F_{XY}$ (Z) and $F_{\Delta H}$ as a function of power and rod position. The $F_{XY}$ (Z) is employed in determining peak linear power density as a function of elevation which is used in the overpower protection system and the LOCA surveillance system (See Section 7). The $F_{\Delta H}$ is employed in the DNBR protection system (See Section 4.4). The following discussion describes the method by which, first, $F_{XY}$ (Z) is obtained and, secondly, $F_{\Delta H}$ is obtained.

The maximum linear power density protection and surveillance systems continuously determine the peak KW/ft as a function of core elevation from the measured core power, axial power distribution, and elevation dependent radial peaking factor. The elevation dependent radial peaking factor is also dependent on the measured rod positions and core power level. Asymptotic $F_{XY}$ (Z) for rodded and unrodded core configurations (ARO, D in, D+C in, D+C+B in) are determined along with the associated power dependence for each configuration during the core design and form part of the input to the KW/ft protection and surveillance systems. As described in more detail in Chapter 7, the composite core $F_{XY}$ (Z) is formed from the asymptotic $F_{XY}$ (Z) for the unrodded and various rodded configurations, the known power dependence for each configuration, and the measured core power and rod positions. The radial peaking factors at selected axial elevations are routinely verified by incore measurements using the moveable detector system as described in specification 3/4.3.4.2 of the Technical Specifications and may be updated at various times throughout the cycle to take advantage of improved margin to core limits due to burnup flattening.

Allowance for the total error in the protection system input parameters is included in the determination of the protection system setpoints as described in Chapter 7.

Increasing $F^N_{\Delta H}$ with decreasing power and increasing control bank insertion is permitted by the DNB protection setpoints as described in Section 4.4. This includes the radial power shape changes with control rods inserted deeper than the insertion limits.

The allowance permitted for increased $F^N_{\Delta H}$ due to decreased power is of the form:

$$F^N_{\Delta H}(\text{Relative Power, Rod Positions}) = F^{HFP}_{\Delta H} (\text{Rod Positions}) [1 + C(1 - P)]$$

Where P is the relative core power and C, the power correction constant, is conservatively determined for each cycle. A value of C=0.10 is typical for most first cycles.

The allowance permitted for increased $F^N_{\Delta H}$ due to increased rod insertion is shown in figure 4.3-46 for full power first cycle operation. The normal operation design basis full power $F_{\Delta H}$ is 1.435 without uncertainty allowance, which is used for establishing acceptable control rod patterns and control bank sequencing. Similarly fuel loading patterns for each cycle are selected with consideration of this design criterion. The worst full power values of $F^N_{\Delta H}$ for possible rod configurations occurring in normal operation are used in verifying that this criterion is met. Typical radial factors are given in Table 4.3-2 and the radial power distributions are shown in Figures 4.3-6 through 4.3-11. The worst normal operation values generally occur when the rods are assumed to be at their insertion limits. However, the worst abnormal $F^N_{\Delta H}$ values for rod bank insertions below their insertion limits are also used in verifying the rod position dependence to ensure $F^N_{\Delta H}$ conservatism during Condition II events. The effect of axial power shape variations on $F^N_{\Delta H}$ although small, are also considered. These limits are taken as input to the thermalhydraulic design basis as described in Section 4.4.

When a situation is possible in normal operation which could result in local power densities in excess of those assumed as the pre-condition for a subsequent hypothetical accident, but which would not itself cause fuel failure, administrative controls and alarms are provided for returning the core to a safe condition.

These alarms are:

1. Axial Flux Difference Monitor,
2. Rod Insertion Limit Monitor,
3. Rod Position Deviation Monitor,
4. Quadrant Power Tilt Ratio Monitor, and
5. High kw/ft.

Alarms 1, 2 and 3 are generated by the plant process computer, 4 by the Nuclear Instrumentation System, and 5 by the Reactor Protection System. Each of these alarms is provided to assist the operator in maintaining the plant within the base assumptions of the accident analyses, i.e., 1) alarms when the operation of the plant has been outside of the CAOC target band for more than one hour of penalty deviation minutes or immediately if power is above 90 percent RTP. Alarm; 2) assists the operator in keeping the control rods 10 steps above the insertion limits (a DNB and $F_Q(z)$ assumption); 3) alarms when rod to rod deviation in a bank is greater than 15 steps indicated, thus keeping control rod alignment reasonably tight and prevents significant reductions in DNBR. Alarm 4) (tilts of greater than 2 percent) assists the operator in maintaining symmetric power distributions and thus minimizes impact on $F^N_{\Delta H}$ and $F_Q(Z)$. Finally alarm 5) notifies the operator when kw/ft, power generation, is getting abnormally high (within 10 percent of the limiting kw/ft) and thus allows sufficient time for power reduction, rod motion (withdrawal) or other actions to reduce the power generation at the affected core elevation. These alarms are described in more detail in Chapter 7.

The appropriate hot channel factors, $F_Q^N$ and $F_{\Delta H}^N$, for peak local power density and for DNB analysis at full power are the values given in Table 4.3-2 and addressed in specifications 3/4.2.1 and 3/4.2.2 of the Technical Specifications.

$F_Q$ can be increased with decreasing power as noted in specification 3/4.2.1 of the Technical Specifications.

4.3.2.2.7 Experimental Verification of Power Distribution Analysis

This subject is discussed in depth in Reference [2]. A summary of this report is given below. It should be noted that power distribution related measurements are incorporated into the evaluation of calculated power distribution using the INCORE code described in Reference [8]. A detailed description of this code's input and output is included in this reference. The measured vs. calculational comparison is normally performed periodically throughout the cycle lifetime of the reactor as required by specification 2.2.1, Table 2.2-1 notes 2 and 4 of the Technical Specifications.

In a measurement of the heat flux hot channel factor, $F_Q$, with the movable detector system described in Sections 7.7.1 and 4.4.6, the following uncertainties have to be considered:

1. Reproducibility of the measured signal

2. Errors in the calculated relationship between detector current and local flux

3. Errors in the calculated relationship between detector flux and peak rod power some distance from the measurement thimble.

The appropriate allowance for Category I above has been quantified by repetitive measurements made with several inter-calibrated detectors by using the common thimble features of the incore detector syst This sytem allows more than one detector to access any thimble. Errors in

Category 2 above are quantified to the extent possible, by using the fluxes measured at one thimble location to predict fluxes at another location which is also measured. Local power distribution predictions are verified in critical experiments on arrays of rods with simulated guide thimbles, control rods, burnable poisons, etc. These critical experiments provide quantification of errors of types 2 and 3 above.

These procedures are detailed in specification 3/4.2.1 of the Technical Specifications and are followed by relying only upon excore surveillance supplemented by the normal monthly full core map requirement and by computer based alarms on deviation and time of deviation from the allowed flux difference band.

Allowing for fuel densification effects, the average linear power at 3411 MWt is 5.44 KW/ft. From Figure 4.3-21, the conservative upper bound value of normalized local power density, including uncertainty allowances, is 2.32 corresponding to a peak linear power of 12.9 KW/ft at 102 percent power.

To determine reactor protection system setpoints, with respect to power distributions, three categories of events are considered, namely rod control equipment malfunctions, operator errors of commission and operator errors of omission. In evaluating these three categories of events, the core is assumed to be operating within four constraints described above.

The first category comprises uncontrolled rod withdrawal (with rods moving in the normal bank sequence) for full length banks. Also included are motions of the full length banks below their insertion limits, which could be caused, for example, by uncontrolled dilution or primary coolant cooldown. Power distributions were calculated throughout these occurrences assuming short term corrective action, that is, no transient xenon effects were considered to result from the malfunction. The event was assumed to occur from typical normal operating situations which include normal xenon transients. It was further assumed in determining the power distributions that total core power level would be limited by reactor trip to below 118 percent. Since the study is to determine protection limits with respect to power and axial offset, no credit was taken for trip setpoint reduction due to flux difference. Results are given in Figure 4.3-22 in units of KW/ft. The peak power density which can occur in such events, assuming reactor trip at or below 118 percent, is less than that required for centerline melt including uncertainties and densification effects.

The second category, also appearing in Figure 4.3-22, assumes that the operator mispositions the full length rod bank in violation of the insertion limits and creates short term conditions not included in normal operating conditions.

The third category assumes that the operator fails to take action to correct a flux difference violation. The results shown on Figure 4.3-23 are $F_Q$ multiplied by 102 percent power which includes a 2 percent allowance for calorimetric error. The figure shows that provided the assumed error in operation does not continue for a period which is long compared to the xenon time constant, the peak linear power does not exceed 18 KW/ft including the above factors.

Analyses of possible operating power shapes show that the appropriate hot channel factors $F_Q$ and $F^N_{\Delta H}$ for peak local power density and for DNB analysis at full power are the values given in Table 4.3-2 and addressed in specifications 3/4.2.2 and 3/4.2.3 of the Technical Specifications.

$F_Q$ can be increased with decreasing power as shown in specification 3/4.2.2 of the Technical Specifications. Increasing $F^N_{\Delta H}$ with decreasing power is permitted by the DNB protection setpoints and allows radial power shape changes with rod insertion to the insertion limits as described in Section 4.4.4.3. The allowance for increased $F^N_{\Delta H}$ permitted is $F^N_{\Delta H} = 1.55\ (1 + 0.3\ (1-P))$. This becomes a design basis criterion which is used for establishing acceptable control rod patterns and control bank sequencing. Likewise fuel loading patterns for each cycle are selected with consideration of this design criterion. The worst values of $F^N_{\Delta H}$ for possible rod configurations occurring in normal operation are used in verifying that this criterion is met. Typical radial factors and radial power distributions are shown in Figures 4.3-6 through 4.3-11. The worst values generally occur when the rods are assumed to be at their insertion limits. Maintenance of constant axial offset control establishes rod positions which are above the allowed rod insertion limits, thus providing increased margin to the $F^N_{\Delta H}$ criterion. As discussed in Section 3.2 of Reference [9], it has been determined that provided the above conditions 1 through 4 are observed, the Technical Specifications limits, are met. These limits are taken as input to the thermal-hydraulic design basis as described in Section 4.4.4.3.1.

When a situation is possible in normal operation which could result in local power densities in excess of those assumed as the pre-condition for a subsequent hypothetical accident, but which would not itself cause fuel failure, administrative controls and alarms are provided for returning the core to a safe condition.

These alarms are:

1. Axial Flux Difference Monitor,
2. Rod Insertion Limit Monitor,
3. Rod Position Deviation Monitor, and
4. Quadrant Power Tilt Ratio Monitor

Alarms 1, 2 and 3 are generated by the plant process computer, and 4 by the Nuclear Instrumentation System. Each of these alarms is provided to assist the operator in maintaining the plant within the base assumptions of the accident analyses, i.e., 1) alarms when the operation of the plant has been outside of the CAOC target band for more than one hour of penalty deviation minutes or immediately if power is above 90 percent RTP. Alarm 2) assists the operator in keeping the control rods 10 steps above the insertion limits (a DNB and $F_Q(Z)$ assumption), 3) alarms when rod to rod deviation in a bank is greater than 15 steps indicated, thus keeping control rod alignment reasonably tight and prevents significant reductions in DNBR. Alarm 4) (tilts of greater than 2 percent) assists the operator in maintaining symmetric power distributions and thus minimizes impact on $F^N_{\Delta H}$ and $F_Q(Z)$. These alarms are described in more detail in Chapter 7.

4.3.2.2.7 Experimental Verification of Power Distribution Analysis

This subject is discussed in depth in Reference [2]. A summary of this report is given below. It should be noted that power distribution related measurements are incorporated into the evaluation of calculated

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power distribution using the INCORE code described in Reference [8]. A detailed description of this code's input and output is included in this reference. The measured vs. calculational comparison is normally performed periodically throughout the cycle lifetime of the reactor as required by specification 3/4.2.2 of the Technical Specifications.

In a measurement of the heat flux hot channel factor, $F_Q$, with the movable detector system described in Sections 7.7.1 and 4.4.6, the following uncertainties have to be considered:

1. Reproducibility of the measured signal

2. Errors in the calculated relationship between detector current and local flux

3. Errors in the calculated relationship between detector flux and peak rod power some distance from the measurement thimble.

The appropriate allowance for Category I above has been quantified by repetitive measurements made with several inter-calibrated detectors by using the common thimble features of the incore detector system. This

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The accumulated data on power distributions in actual operation is basically of three types:

1. Much of the data is obtained in steady state operation at constant power in the normal operating configuration;

2. Data with unusual values of axial offset are obtained as part of the excore detector calibration exercise which is performed monthly;

3. Special tests have been performed in load follow and other transient xenon conditions which have yielded useful information on power distributions.

These data are presented in detail in References [9],[39]. Figure 4.3-26 contains a summary of measured values of $F_Q$ as a function of axial offset for several plants from these reports.

4.3.2.2.8 Testing

A very extensive series of physics tests is performed on the first core, even though this core is not a prototype design. These tests and the criteria for satisfactory results are described in Chapter 14. Since not all limiting situations can be created at beginning-of-life, the main purpose of the tests is to provide a check on the calculational methods used in the predictions for the conditions of the test. Tests performed at the beginning of each reload cycle are limited to verification of steady state power distributions, on the assumptions that the reload fuel is supplied by the first core designer.

4.3.2.2.9 Monitoring Instrumentation

The adequacy of instrument numbers, spatial deployment, required correlations between readings and peaking factors, calibration and errors are described in References [2], [6], and [9]. The relevant conclusions are summarized here in Sections 4.3.2.2.7 and 4.4.6.

Provided the limitations given in Section 4.3.2.2.6 on control rods moving together in a single bank and control banks sequenced with design overlap, the multi-section excore detector based surveillance system provides adequate online monitoring of power distributions. Further details of specific limits on the observed rod positions and power distributions are given in specifications 3/4.1.3.1, 3/4.1.3.6, 3/4.2.1, and 3/4.2.2 (3/4.2.2 and 3/4.2.3 - Blue) of the Technical Specifications. Descriptions of the systems provided are given in Section 7.7.

4.3.2.3 Reactivity Coefficients

The kinetic characteristics of the reactor core determine the response of the core to changing plant conditions or to operator adjustments made during normal operation, as well as the core response during abnormal or accidental transients. The reactivity coefficients reflect the changes in the neutron multiplication due to varying plant conditions such as power, moderator or fuel temperatures, or less significantly due to a change in pressure or void conditions. Since reactivity coefficients change during the life of the core, ranges of coefficients are employed in transient analysis to determine the response of the plant throughout life. The results of such simulations and the reactivity coefficients used are presented in Chapter 15. The reactivity coefficients are calculated on a corewise basis by radial and axial diffusion theory methods and with nodal analysis methods. The effect of radial and axial power distribution on core average reactivity coefficients is implicit in those calculations and is not significant under normal operating conditions. For example, a skewed xenon distribution which results in changing axial offset by 5 percent changes the moderator and Doppler temperature coefficients by less than 0.01 pcm/$^{o}$F and 0.03 pcm/$^{o}$F respectively. An artificially skewed xenon distribution which results in changing the radial $F_{\Delta H}$ by 3 percent changes the moderator and Doppler temperature coefficients by less than 0.03 pcm/$^{o}$F and 0.001 pcm/$^{o}$F respectively. The spatial effects are accentuated in some transient conditions; for example, in postulated rupture of the main steamline break and rupture of RCCA mechanism housing described in Sections 15.1.5 and 15.4.8, and are included in these analyses.

4.3.2.3.4 Comparison of Calculated and Experimental Reactivity Coefficients

Section 4.3.3 describes the comparison of calculated and experimental reactivity coefficients in detail. Based on the data presented there, the accuracy of the current analytical model is:

$\pm$0.2 percent $\Delta\rho$ for Doppler and power defect

$\pm$2 pcm/$^{o}$F for the moderator coefficient.

Experimental evaluation of the calculated coefficients will be done during the physics startup tests described in Chapter 14.

4.3.2.3.5 Reactivity Coefficients Used in Transient Analysis

Table 4.3-2 gives the limiting values as well as the best estimate values for the reactivity coefficients. The limiting values are used as design limits in the transient analysis. The exact values of the coefficient used in the analysis depend on whether the transient of interest is examined at the BOL or EOL, whether the most negative or the most positive (least negative) coefficients are appropriate, and whether spatial nonuniformity must be considered in the analysis. Conservative values of coefficients, considering various aspects of analysis are used in the transient analysis. This is described in Chapter 15.

The reactivity coefficients shown in Figures 4.3-27 through 4.3-35 are best estimate values calculated for this cycle and apply to the core described in Table 4.3-1. The limiting values shown in Table 4.3-2 are chosen to encompass the best estimate reactivity coefficients, including the uncertainties given in Section 4.3.3.3 over appropriate operating conditions calculated for this cycle and the expected values for the subsequent cycles. The most positive as well as the most negative values are selected to form the design basis range used in the transient analysis. A direct comparison of the best estimate and design limit values shown in Table 4.3-2 can be misleading since in many instances, the most conservative combination of reactivity coefficients is used in the transient analysis even though the extreme coefficients assumed may not simultaneously occur at the conditions of lifetime, power level, temperature and boron concentration assumed in the analysis. The need for a reevaluation of any accident in a subsequent cycle is contingent upon whether or not the coefficients for that cycle fall within the identified range used in the analysis presented in Chapter 15 with due allowance for the calculational uncertainties given in Section 4.3.3.3. Control rod requirements are given in Table 4.3-3 for the core described and for a hypothetical equilibrium cycle since these are markedly different. These latter numbers are provided for information only and their validity in a particular cycle would be an unexpected coincidence.

#### 4.3.2.4 Control Requirements

To ensure the shutdown margin stated in specifications 3/4.1.1.1 and 3/4.1.1.2 of the Technical Specifications under conditions where a cooldown to ambient temperature is required, concentrated soluble boron is added to the coolant. Boron concentrations for several core conditions are listed in Table 4.3-2. For all core conditions including refueling, the boron concentration is well below the solubility limit. The rod cluster control assemblies are employed to bring the reactor to the hot shutdown condition.

The ability to accomplish the shutdown for hot conditions is demonstrated in Table 4.3-3 by comparing the difference between the rod cluster control assembly reactivity available with an allowance for the worst stuck rod with that required for control and protection purposes. The shutdown margin includes an allowance of 10 percent for analytic uncertainties (see Section 4.3.2.4.9). The largest reactivity control requirement appears at the EOL when the moderator temperature coefficient reaches its peak negative value as reflected in the larger power defect.

3. Unprogrammed fluctuations in boron concentration, coolant temperature, or xenon concentration (with rods not exceeding the allowable rod insertion limits),

4. Reactivity ramp rates resulting from load changes.

The allowed full length control bank reactivity insertion is limited at full power to maintain shutdown capability. As the power level is reduced, control rod reactivity requirements are also reduced and more rod insertion is allowed. The control bank position is monitored and the operator is notified by the Rod Insertion Limit Monitor if the limit is approached. The determination of the insertion limit uses conservative xenon distributions and axial power shapes. In addition, the rod cluster control assembly withdrawal pattern determined from these analyses is used in determining power distribution factors and in determining the maximum worth of an inserted rod cluster control assembly ejection accident. Specification 3/4.1.3.6 of the Technical Specifications provides the rod insertion limits.

Power distribution, rod ejection and rod misalignment analyses are based on the arrangement of the shutdown and control groups of the rod cluster control assemblies shown in Figure 4.3-36. All shutdown rod cluster control assemblies are withdrawn before withdrawal of the control banks is initiated. In going from zero to 100 percent power, control banks A, B, C and D are withdrawn sequentially. The limits of rod positions and further discussion on the basis for rod insertion limits are provided in the above noted Technical Specifications.

4.3.2.4.13 Reactor Coolant Temperature

Reactor coolant (or moderator) temperature control has added flexibility in reactivity control of the Westinghouse PWR. This feature takes advantage of the negative moderator temperature coefficient inherent in a PWR to:

1. Maximize return to power capabilities.

4.3.2.4.15 Peak Xenon Startup

Compensation for the peak xenon buildup is accomplished using the boron control system. Startup from the peak xenon condition is accomplished with a combination of rod motion and boron dilution. The boron dilution may be made at any time, including during the shutdown period, provided the shutdown margin is maintained.

4.3.2.4.16 Load Follow Control and Xenon Control

During load follow maneuvers, power changes are accomplished using control rod motion and dilution or boration by the boron system as required. Control rod motion is limited by the control rod insertion limits on full length rods as provided in specification 3/4.1.3.6 of the Technical Specifications and discussed in Sections 4.3.2.4.12 and 4.3.2.4.13. The power distribution is maintained within acceptable limits through the location of the full length rod bank. Reactivity changes due to the changing xenon concentration can be controlled by rod motion and/or changes in the soluble boron concentration.

Late in cycle life, extended load follow capability is obtained by augmenting the limited boron dilution capability at low soluble boron concentrations by temporary moderator temperature reductions.

Rapid power increases (5 percent/min) from part power during load follow operation are accomplished with a combination of rod motion, moderator temperature reduction, and boron dilution. Compensation for the rapid power increase is accomplished initially by a combination of rod withdrawal and moderator temperature reduction. As the slower boron dilution takes affect after the initial rapid power increase, the moderator temperature returns to the programmed value.

4.3.2.4.17 Burnup

Control of the excess reactivity for burnup is accomplished using soluble boron and/or burnable poison. The boron concentration must be limited during operating conditions to ensure the moderator temperature will be achieved with control rods above the insertion limit set by shutdown and other considerations (see specification 3/4.1.3.6 of the Technical Specifications). Early in the cycle there may also be a withdrawal limit at low power to maintain a negative moderator temperature coefficient. For the reference first core design described in this chapter, however, no such withdrawal limit is required.

Ejected rod worths are given in Section 15.4.8 for several different conditions.

Allowable deviations due to misaligned control rods are noted in specification 3/4.1.3.1 of the Technical Specifications.

A representative calculation for two banks of control rods withdrawn simultaneously (rod withdrawal accident) is given in Figure 4.3-37.

Calculation of control rod reactivity worth versus time following reactor trip involves both control rod velocity and differential reactivity worth. The rod position versus time of travel after rod release normalized to "Distance to Top of Dashpot" and Drop Time to Top of Dashpot" is given in Figure 4.3-38 for hybrid RCC material. For nuclear design purposes, the reactivity worth versus rod position is calculated by a series of steady state calculations at various control rod positions assuming all rods out of the core as the initial position in order to minimize the initial reactivity insertion rate. Also to be conservative, the rod of highest worth is assumed stuck out of the core and the flux distribution (and thus reactivity importance) is assumed to be skewed to the bottom of the core. The result of these calculations is shown on Figure 4.3-39.

The shutdown groups provide additional negative reactivity to assure an adequate shutdown margin. Shutdown margin is defined as the amount by which the core would be subcritical at hot shutdown if all rod cluster control assemblies are tripped, but assuming that the highest worth assembly remains fully withdrawn and no changes in xenon or boron take

Axial calculations are used to determine differential control rod worth curves (reactivity versus rod insertion) and axial power shapes during steady state and transient xenon conditions (flyspeck curve). Group constants and the radial buckling used in the axial calculation are obtained from the PANDA radial calculation, in which group constants in annular rings representing the various material regions in the X-Y plane are homogenized by flux-volume weighting.

Validation of the spatial codes for calculating power distributions involves the use of incore and excore detectors and is discussed in Section 4.3.2.2.7.

Based on comparison with measured data it is estimated that the accuracy of current analytical methods is:

$\pm$ 0.2 percent $\Delta\rho$ for Doppler defect
$\pm$ 2 x $10^{-5}$ $\Delta\rho/^{o}F$ for moderator coefficient
$\pm$ 50 ppm for critical boron concentration with depletion
$\pm$ 3 percent for power distributions
$\pm$ 0.2 percent $\Delta\rho$ for rod bank worth
$\pm$ 4 pcm/step for differential rod worth
$\pm$ 0.5 pcm/ppm for boron worth
$\pm$ 0.1 percent $\Delta\rho$ for moderator defect

## 4.3.4 CHANGES

The design methods for the criticality of fuel assemblies outside the reactor now uses the AMPX/KENO ORNL system of codes as described in Section 4.3.2.6.

The design methods for the nuclear analysis of the core now use both TURTLE (Reference 10) and PALADON (Reference 38) for multi-dimensional analyses.

annular rings representing the various material regions in the X-Y plane are homogenized by flux-volume weighting.

Validation of the spatial codes for calculating power distributions involves the use of incore and excore detectors and is discussed in Section 4.3.2.2.7.

Based on comparison with measured data it is estimated that the accuracy of current analytical methods is:

- $\pm$ 0.2 percent $\Delta\rho$ for Doppler defect
- $\pm$ 2 x $10^{-5}\Delta\rho/^{o}F$ for moderator coefficient
- $\pm$ 50 ppm for critical boron concentration with depletion
- $\pm$ 3 percent for power distributions
- $\pm$ 0.2 percent $\Delta\rho$ for rod bank worth
- $\pm$ 4 pcm/step for differential rod worth
- $\pm$ 0.5 pcm/ppm for boron worth
- $\pm$ 0.1 percent $\Delta\rho$ for moderator defect

## 4.3.4 CHANGES

The design methods for the criticality of fuel assemblies outside the reactor now uses the AMPX/KENO system of codes as described in Section 4.3.2.6.

The design methods for the nuclear analysis of the core now use both TURTLE (Reference 10) and PALADON (Reference 38) for multi-dimensional analyses.

The fuel assembly is of an improved mechanical design which employes a slightly reduced fuel rod clad outer diameter and pellet diameter while retaining the same fuel rod pitch. Another feature is the incorporation of Zircaloy spacer grids for all buth the top and bottom spacer grid locations which will continue to be fabricated from Inconel. The physics characteristics are provided throughout Section 4.3.