APPENDIX C

Technical Specification Change 001

Justification for change:

Rod drop time requirement is increased for consistency with the assumption used throughout the revised DCD Chapter 15 Safety Analysis.

SR 3.1.4.3 based on DCD Revision 19 would be changed from:

SR 3.1.4.3	- NOTE - Not applicable to GRCAs. Verify rod drop time of each rod, from the fully withdrawn position, is ≤ 2.47 seconds from the beginning of decay of stationary gripper coil voltage to dashpot entry, with: a. $T_{avg} \geq 500°F$, and b. All reactor coolant pumps operating.	Prior to reactor criticality after each removal of the reactor head, and after each earthquake requiring plant shutdown

Marked up version:

SR 3.1.4.3	- NOTE - Not applicable to GRCAs. Verify rod drop time of each rod, from the fully withdrawn position, is ≤ 2.7 seconds from the beginning of decay of stationary gripper coil voltage to dashpot entry, with: a. $T_{avg} \geq 500°F$, and b. All reactor coolant pumps operating.	Prior to reactor criticality after each removal of the reactor head, and after each earthquake requiring plant shutdown

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To read:

SR 3.1.4.3	-- **- NOTE -** Not applicable to GRCAs. -- Verify rod drop time of each rod, from the fully withdrawn position, is $\leq$ 2.7 seconds from the beginning of decay of stationary gripper coil voltage to dashpot entry, with: a. $T_{avg} \geq 500°F$, and b. All reactor coolant pumps operating.	Prior to reactor criticality after each removal of the reactor head, and after each earthquake requiring plant shutdown

Technical Specification Change 002

Justification for change

Consistent with the revised core, fuel and safety analysis, LCO 3.2.1 is revised to reflect a Constant Axial Offset Control technical specification limit with $F_Q(z)$ measurements following the standard specification description in NUREG-1431 Rev. 3.

LCO 3.2.1 based on DCD Revision 19 will be changed from:

3.2 POWER DISTRIBUTION LIMITS

3.2.1 Heat Flux Hot Channel Factor ($F_Q(Z)$) (F_Q Methodology)

Marked up version:

3.2 POWER DISTRIBUTION LIMITS

3.2.1 Heat Flux Hot Channel Factor ($F_Q(Z)$) (Constant Axial Offset Control (CAOC) -W(z) Methodology)

Deleted: F_Q Methodology

To read:

3.2 POWER DISTRIBUTION LIMITS

3.2.1 Heat Flux Hot Channel Factor ($F_Q(Z)$) (Constant Axial Offset Control (CAOC) -W(z) Methodology)

Technical Specification Change 003

Justification for change

Consistent with the revised core, fuel and safety analysis, LCO 3.2.1 is revised to reflect a Constant Axial Offset Control technical specification limit with $F_Q(z)$ measurements following the standard specification description in NUREG-1431 Rev. 3.

Actions for LCO 3.2.1 based on DCD Revision 19 will be changed from:

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. -------------------------------- - **NOTE** - Required Action B.4 shall be completed whenever this Condition is entered. -------------------------------- $F_Q^W(Z)$ not within limits.	B.1	Reduce AFD limits ≥ 1% for each 1% $F_Q^W(Z)$ exceeds limit.	4 hours
	<u>AND</u>		
	B.2	Reduce Power Range Neutron Flux – High trip setpoints ≥ 1% for each 1% that the maximum allowable power of the AFD limits is reduced.	72 hours
	<u>AND</u>		
	B.3	Reduce Overpower ΔT trip setpoints ≥ 1% for each 1% that the maximum allowable power of the AFD limits is reduced.	72 hours
	<u>AND</u>		
	B.4	Perform SR 3.2.1.1 and SR 3.2.1.2.	Prior to increasing THERMAL POWER above the maximum allowable power of the AFD limits
C. Required Action and associated Completion Time not met.	C.1	Be in MODE 2.	6 hours

Marked up version:

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. -------------------------------- **- NOTE -** Required Action B.4 shall be completed whenever this Condition is entered. -------------------------------- $F_Q^W(Z)$ not within limits.	B.1	Reduce THERMAL POWER ≥ 1% for each 1% $F_Q^W(Z)$ exceeds limit.	4 hours
	AND		
	B.2	Reduce Power Range Neutron Flux – High trip setpoints ≥ 1% for each 1% $F_Q^W(Z)$ exceeds limit.	72 hours
	AND		
	B.3	Reduce Overpower ΔT trip setpoints ≥ 1% for each 1% $F_Q^W(Z)$ exceeds limit..	72 hours
	AND		
	B.4	Perform SR 3.2.1.1 and SR 3.2.1.2.	Prior to increasing THERMAL POWER above the limit of Required Action B.1
C. Required Action and associated Completion Time not met.	C.1	Be in MODE 2.	6 hours

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To read:

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. ------------------------------- **- NOTE -** Required Action B.4 shall be completed whenever this Condition is entered. ------------------------------- $F_Q^W(Z)$ not within limits.	B.1	Reduce THERMAL POWER ≥ 1% for each 1% $F_Q^W(Z)$ exceeds limit.	4 hours
	AND		
	B.2	Reduce Power Range Neutron Flux – High trip setpoints ≥ 1% for each 1% $F_Q^W(Z)$ exceeds limit.	72 hours
	AND		
	B.3	Reduce Overpower ΔT trip setpoints ≥ 1% for each 1% $F_Q^W(Z)$ exceeds limit.	72 hours
	AND		
	B.4	Perform SR 3.2.1.1 and SR 3.2.1.2.	Prior to increasing THERMAL POWER above the limit of Required Action B.1
C. Required Action and associated Completion Time not met.	C.1	Be in MODE 2.	6 hours

---

Technical Specification Change 004

Justification for change

Consistent with the revised core, fuel and safety analysis, TS 3.2.3 is revised to reflect a Constant Axial Offset Control technical specification limit with $F_Q(z)$ measurements following the standard specification description in NUREG-1431 Rev. 3.

TS 3.2.3 based on DCD Revision 19 will be changed from:

3.2 POWER DISTRIBUTION LIMITS

3.2.3 AXIAL FLUX DIFFERENCE (AFD) (Relaxed Axial Offset Control (RAOC) Methodology)

LCO 3.2.3 The AFD in %-flux-difference units shall be maintained within the limits specified in the COLR.

- NOTE -

The AFD shall be considered outside limits when two or more OPERABLE excore channels indicate AFD to be outside limits.

APPLICABILITY: MODE 1 with THERMAL POWER ≥ 50% RTP and with the On-Line Power Distribution Monitoring System (OPDMS) inoperable.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. AFD not within limits.	A.1 Reduce THERMAL POWER to < 50% RTP.	30 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.3.1	Verify AFD within limits for each OPERABLE excore channel.	7 days

Marked up version:

3.2 POWER DISTRIBUTION LIMITS

3.2.3 AXIAL FLUX DIFFERENCE (AFD) (Constant Axial Offset Control (CAOC) Methodology)

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LCO 3.2.3 The AFD:

a. Shall be maintained within the target band about the target flux difference. The target band is specified in the COLR.

b. May deviate outside the target band with THERMAL POWER <90% RTP but ≥ 50% RTP, provided AFD is within the acceptable operation limits and cumulative penalty deviation time is ≤ 1 hour during the previous 24 hours. The acceptable operation limits are specified in the COLR.

c. May deviate outside the target band with THERMAL POWER < 50% RTP.

Deleted: in %-flux-difference units shall be maintained within the limits specified in the COLR.

- NOTES -

1. The AFD shall be considered outside the target band when two or more OPERABLE excore channels indicate AFD to be outside the target band.
2. With THERMAL POWER ≥ 50% RTP, penalty deviation time shall be accumulated on the basis of a 1 minute penalty deviation for each 1 minute of power operation with AFD outside the target band.
3. With THERMAL POWER < 50% RTP and > 15% RTP, penalty deviation time shall be accumulated on the basis of a 0.5 minute penalty deviation for each 1 minute of power operation with AFD outside the target band.
4. A total of 16 hours of operation may be accumulated with AFD outside the target band without penalty deviation time during surveillance of power range channels in accordance with SR 3.3.1.5, provided AFD is maintained within acceptable operation limits

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APPLICABILITY: MODE 1 with THERMAL POWER > 15% RTP and with the On-Line Power Distribution Monitoring System (OPDMS) inoperable.

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ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. THERMAL POWER ≥ 90% RTP AND AFD not within the target band	A.1 Restore AFD to within target band	15 minutes
B. Required Action and associated Completion Time of Condition A not met.	B.1 Reduce THERMAL POWER to < 90% RTP.	15 minutes
C. ----------NOTE------------- Required Action C.1 must be completed whenever Condition C is entered. -------------------------------- THERMAL POWER < 90% and ≥ 50% RTP with cumulative penalty deviation time > 1 hour during the previous 24 hours. OR THERMAL POWER < 90% and ≥ 50% with AFD not within the acceptable operation limits.	C.1 Reduce THERMAL POWER to < 50% RTP.	30 minutes
D. Required Action and associated Completion Time for Condition C not met.	D.1 Reduce THERMAL POWER to < 15% RTP.	9 hours

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.3.1	Verify AFD within limits for each OPERABLE excore channel.	7 days
SR 3.2.3.2	Update the target flux difference.	Once within 31 EFPD after each refueling AND 31 EFPD thereafter
	-- - NOTE - The initial target flux difference after each refueling may be determined from design predictions. --	
SR 3.2.3.3	Determine, by measurement, the target flux difference.	Once within 31 EFPD after each refueling AND 92 EFPD thereafter

---

To read:

3.2 POWER DISTRIBUTION LIMITS

3.2.3 AXIAL FLUX DIFFERENCE (AFD) (Constant Axial Offset Control (CAOC) Methodology)

LCO 3.2.3 The AFD:

a. Shall be maintained within the target band about the target flux difference. The target band is specified in the COLR.

b. May deviate outside the target band with THERMAL POWER <90% RTP but ≥ 50% RTP, provided AFD is within the acceptable operation limits and cumulative penalty deviation time is ≤ 1 hour during the previous 24 hours. The acceptable operation limits are specified in the COLR.

c. May deviate outside the target band with THERMAL POWER < 50% RTP.

- NOTES -

1. The AFD shall be considered outside the target band when two or more OPERABLE excore channels indicate AFD to be outside the target band.

2. With THERMAL POWER ≥ 50% RTP, penalty deviation time shall be accumulated on the basis of a 1 minute penalty deviation for each 1 minute of power operation with AFD outside the target band.

3. With THERMAL POWER < 50% RTP and > 15% RTP, penalty deviation time shall be accumulated on the basis of a 0.5 minute penalty deviation for each 1 minute of power operation with AFD outside the target band.

4. A total of 16 hours of operation may be accumulated with AFD outside the target band without penalty deviation time during surveillance of power range channels in accordance with SR 3.3.1.5, provided AFD is maintained within acceptable operation limits..

APPLICABILITY: MODE 1 with THERMAL POWER > 15% RTP and with the On-Line Power Distribution Monitoring System (OPDMS) inoperable.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. THERMAL POWER ≥ 90% RTP AND AFD not within the target band.	A.1	Restore AFD to within target band.	15 minutes
B. Required Action and associated Completion Time of Condition A not met.	B.1	Reduce THERMAL POWER to < 90% RTP.	15 minutes
C. ----------NOTE------------ Required Action C.1 must be completed whenever Condition C is entered. -------------------------------- THERMAL POWER < 90% and ≥ 50% RTP with cumulative penalty deviation time > 1 hour during the previous 24 hours. OR THERMAL POWER < 90% and ≥ 50% with AFD not within the acceptable operation limits.	C.1	Reduce THERMAL POWER to < 50% RTP.	30 minutes
D. Required Action and associated Completion Time for Condition C not met.	D.1	Reduce THERMAL POWER to < 15% RTP.	9 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.2.3.1	Verify AFD within limits for each OPERABLE excore channel.	7 days
SR 3.2.3.2	Update the target flux difference.	Once within 31 EFPD after each refueling <u>AND</u> 31 EFPD thereafter
SR 3.2.3.3	-------------------------------- **- NOTE -** The initial target flux difference after each refueling may be determined from design predictions. -------------------------------- Determine, by measurement, the target flux difference.	Once within 31 EFPD after each refueling <u>AND</u> 92 EFPD thereafter

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.1 SHUTDOWN MARGIN (SDM)

BASES

BACKGROUND

According to GDC 26 (Ref. 1) the reactivity control systems must be redundant and capable of holding the reactor core subcritical when shutdown under cold conditions. Maintenance of the SDM ensures that postulated reactivity events will not damage the fuel.

SDM requirements provide sufficient reactivity margin to assure that acceptable fuel design limits will not be exceeded for normal shutdown and anticipated operational occurrences (AOOs). As such, the SDM defines the degree of subcriticality that would be obtained immediately following the insertion or scram of all Rod Cluster Control Assemblies (RCCAs), assuming that the single rod cluster assembly of highest reactivity worth is fully withdrawn.

The system design requires that two independent reactivity control systems be provided, and that one of these systems be capable of maintaining the core subcritical under cold conditions. These requirements are provided by the use of movable control assemblies and soluble boric acid in the Reactor Coolant System (RCS). The Plant Control System (PLS) can compensate for the reactivity effects of the fuel and water temperature changes accompanying power level changes over the range from full load to no load. In addition, the PLS, together with the boration system, provides the SDM during power operation and is capable of making the core subcritical rapidly enough to prevent exceeding acceptable fuel damage limits, assuming that the rod of highest reactivity worth remains fully withdrawn. The soluble boron system can compensate for fuel depletion during operation and xenon burnout reactivity changes and maintain the reactor subcritical under cold conditions.

During power operation, SDM is calculated and monitored by the Online Power Distribution Monitoring System (OPDMS) and controlled by operating with RCCAs sufficiently withdrawn to meet the SDM requirement. When the OPDMS is inoperable, SDM control is ensured by operating within the limits of LCO 3.1.5 "Shutdown Bank Insertion Limits," and LCO 3.1.6, "Control Bank Insertion Limits." When the unit is in the shutdown and refueling modes, the SDM requirements are met by adjustments to the RCS boron concentration.

APPLICABLE SAFETY ANALYSES

The minimum required SDM is assumed as an initial condition in safety analyses. The safety analyses (Ref. 2) establish an SDM that ensures that specified acceptable fuel design limits are not exceeded for normal operation and AOOs, with the assumption of the highest worth rod stuck out on scram. For MODE 5, the primary safety analysis that relies on the SDM limits is the boron dilution analysis.

The acceptance criteria for the SDM requirements are that specified acceptable fuel design limits are maintained. This is done by ensuring that:

a. The reactor can be made subcritical from all operating conditions, transients, and Design Basis Events;

b. The reactivity transients associated with postulated accident conditions are controllable within acceptable limits (departures from nucleate boiling ratio (DNBR), fuel centerline temperature limits for AOOs, and must not exceed applicable limits (reference 4) for the rod ejection accident); and

Deleted: ≤ 280 cal/gm energy deposition

c. The reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

The most limiting accidents for the SDM requirements are based on a main steam line break (SLB) and inadvertent opening of a steam generator (SG) relief or safety valve, as described in the accident analyses (Ref. 2). The increased steam flow in the main steam system causes an increased energy removal from the affected SG, and consequently the RCS. This results in a reduction of the reactor coolant temperature. The resultant coolant shrinkage causes a reduction in pressure. In the presence of a negative moderator temperature coefficient (MTC), this cooldown causes an increase in core reactivity. The positive reactivity addition from the moderator temperature decrease will terminate when the affected SG boils dry, thus terminating RCS heat removal and cooldown. Following the SLB or opening of an SG relief or safety valve, a post trip return to power may occur; however, no fuel damage occurs as a result of the post trip return to power, and the THERMAL POWER does not violate the Safety Limit (SL) requirement of SL 2.1.1.

In addition to the limiting SLB and inadvertent opening of an SG relief or safety valve transients, the SDM requirement must also protect against:

a. Inadvertent boron dilution;

b. An uncontrolled rod withdrawal from subcritical or low power condition;

APPLICABLE SAFETY ANALYSES (continued)

c. Rod ejection.

Each of these events is discussed below.

In the boron dilution analysis, the required SDM defines the reactivity difference between an initial subcritical boron concentration and the corresponding critical boron concentration. These values, in conjunction with the configuration of the RCS and the assumed dilution flow rate, directly affect the results of the analysis. This event is most limiting when critical boron concentrations are highest.

The uncontrolled rod withdrawal transient is terminated by a high neutron flux trip. Power level, RCS pressure, linear heat rate, and the DNBR do not exceed allowable limits.

The ejection of a control rod rapidly adds reactivity to the reactor core, causing both the core power level and heat flux to increase with corresponding increases in reactor coolant temperatures and pressure. The ejection of a rod also produces a time-dependent redistribution of core power.

SDM satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii). Even though it is not directly observed from the main control room, SDM is considered an initial condition process variable because it is periodically monitored to provide assurance that the unit is operating within the bounds of accident analysis assumptions.

LCO

SDM is a core design condition that can be ensured during operation through calculations by the OPDMS and RCCA positioning and through the soluble boron concentration.

The SLB and the boron dilution accidents (Ref. 2) are the most limiting analyses that establish the SDM value of the LCO. For SLB accidents, if the LCO is violated, there is a potential to exceed the DNBR limit and to exceed 10 CFR 50.34 limits (Ref. 3). For the boron dilution accident, if the LCO is violated, the minimum required time assumed for automatic action to terminate dilution may no longer be applicable.

BASES

APPLICABILITY

In MODE 2 with k_{eff} < 1.0, and in MODES 3, 4, and 5, the SDM requirements are applicable to provide sufficient negative reactivity to meet the assumptions of the safety analyses discussed above. In MODE 6, the shutdown reactivity requirements are given in LCO 3.9.1, "Boron Concentration." In MODES 1 and 2, SDM is ensured by complying with LCO 3.1.5, "Shutdown Bank Insertion Limits," and LCO 3.1.6, "Control Bank Insertion Limits."

ACTIONS

A.1

If the SDM requirements are not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. It is assumed that boration will be continued until the SDM requirements are met.

In the determination of the required combination of boration flow rate and boron concentration, there is no unique requirement that must be satisfied. Since it is imperative to raise the boron concentration of the RCS as soon as possible, the boron concentration should be a concentrated solution. The operator should begin boration with the best source available for the plant conditions.

In determining the boration flow rate, the time in core life must be considered. For instance, the most difficult time in core life to increase the RCS boron concentration is at hot shutdown conditions when boron concentration is highest at 1502 ppm. Assuming that a value of 1.0% Δk/k must be recovered and the boration flow rate is 100 gpm, it is possible to increase the boron concentration of the RCS by 111 ppm in approximately 21 minutes utilizing boric acid solution having a concentration of 4375 ppm. If a boron worth of 9 pcm/ppm is assumed, this combination of parameters will increase the SDM by 1.0% Δk/k. These boration parameters of 100 gpm and 4375 ppm represent typical values and are provided for the purpose of offering a specific example.

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1

In MODES 1 and 2 with $k_{eff} \geq 1.0$, SDM is verified by observing that the requirements of LCO 3.1.5 and LCO 3.1.6 are met. In the event that an RCCA is known to be untrippable, however, SDM verification must account for the worth of both the untrippable RCCA as well as another RCCA of maximum worth.

In MODES 3, 4, and 5, the SDM is verified by performing a reactivity balance calculation, considering at least the listed reactivity effects:

a. RCS boron concentration;

b. RCCA and GRCA position;

c. RCS average temperature;

d. Fuel burnup based on gross thermal energy generation;

e. Xenon concentration;

f. Samarium concentration; and

g. Isothermal Temperature Coefficient (ITC).

Using the ITC accounts for Doppler reactivity in this calculation because the reactor is subcritical and the fuel temperature will be changing at the same rate as the RCS.

The Frequency of 24 hours is based on the generally slow change in required boron concentration and the low probability of an accident occurring without the required SDM. This allows time for the operator to collect the required data, which includes performing a boron concentration analysis, and complete the calculation.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26.
2. Chapter 15, "Accident Analysis."
3. 10 CFR 50.34.
4. NUREG-0800, Standard Review Plan (SRP), Section 4.2, Revision 3, "Fuel System Design," (Appendix B).

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B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1 Heat Flux Hot Channel Factor ($F_Q(Z)$) (Constant Axial Offset Control (CAOC) W(Z) Methodology)

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BASES

BACKGROUND

The purpose of the limits on the values of $F_Q(Z)$ is to limit the local (i.e., pellet) peak power density. The value of $F_Q(Z)$ varies along the axial height (Z) of the core.

$F_Q(Z)$ 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 fuel rod dimensions. Therefore, $F_Q(Z)$ is a measure of the peak fuel pellet power within the reactor core.

During power operation with the On-line Power Distribution Monitoring System (OPDMS) inoperable, the global power distribution is limited by LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," which are directly and continuously measured process variables. These LCOs along with LCO 3.1.6, "Control Bank Insertion Limits," maintain the core limits on power distributions on a continuous basis.

$F_Q(Z)$ varies with fuel loading patterns, control bank insertion, fuel burnup, and changes in axial power distribution.

With the OPDMS OPERABLE, peak kw/ft (Z) (which is proportional to $F_Q(Z)$) is measured continuously. With the OPDMS inoperable, $F_Q(Z)$ is measured periodically using the incore detector system. These measurements are generally taken with the core at or near steady state conditions.

With the measured three dimensional power distributions, it is possible to derive a measured value for $F_Q(Z)$ with the OPDMS inoperable. However, because this value represents a steady state condition, it does not include the variations in the value of $F_Q(Z)$ which are present during a nonequilibrium situation such as load following.

To account for these possible variations, the steady state value of $F_Q(Z)$ is adjusted by an elevation dependent factor to account for the calculated worst case transient conditions.

Core monitoring and control under non-equilibrium conditions and the OPDMS inoperable are accomplished by operating the core within the limits of the appropriate LCOs, including the limits on AFD, QPTR, and control rod insertion.

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BASES

APPLICABLE SAFETY ANALYSES

This LCO precludes core power distributions that violate the following fuel design criteria:

a. During a large break loss of coolant accident (LOCA), the peak cladding temperature must not exceed a limit of 2200°F (Ref. 1);

b. During a loss of forced reactor coolant flow accident, there must be at least a 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a departure from nucleate boiling (DNB) condition;

c. During an ejected rod accident, the energy deposition to the fuel must not exceed applicable limits (Ref. 2); and

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d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 3).

Limits on $F_Q(Z)$ ensure that the value of the initial total peaking factor assumed in the accident analyses remains valid. Other criteria must also be met (e.g., maximum cladding oxidation, maximum hydrogen generation, coolable geometry, and long term cooling). However, the peak cladding temperature is typically most limiting.

$F_Q(Z)$ limits assumed in the LOCA analysis are typically limiting (i.e., lower than) relative to the $F_Q(Z)$ assumed in safety analyses for other postulated accidents. Therefore, this LCO provides conservative limits for other postulated accidents.

$F_Q(Z)$ satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The Heat Flux Hot Channel Factor, $F_Q(Z)$, shall be limited by the following relationships:

$$F_Q(Z) \le \frac{CFQ}{P} \qquad \text{for } P > 0.5$$

$$F_Q(Z) \le \frac{CFQ}{0.5} \qquad \text{for } P \le 0.5$$

where: CFQ is the $F_Q(Z)$ limit at RTP provided in the COLR,

$$P = \frac{\text{THERMAL POWER}}{\text{RTP}}$$

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BASES

The actual values of CFQ are given in the COLR; however, CFQ is normally a number on the order of 2.60. For the AP1000, the normalized $F_Q(Z)$ as a function of core height is 1.0.

For Constant Axial Offset Control (CAOC) operation, $F_Q(Z)$ is approximated by $F_Q^C(Z)$ and $F_Q^W(Z)$. Thus, both $F_Q^C(Z)$ and $F_Q^W(Z)$ must meet the preceding limits on $F_Q(Z)$.

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An $F_Q^C(Z)$ evaluation requires obtaining an incore flux map in MODE 1. From the incore flux map results the measured value of $F_Q(Z)$, called $F_Q^M(Z)$ is obtained. Then,

$$F_Q^C(Z) = F_Q^M(Z) * F_Q^{MU}(Z)$$

where $F_Q^{MU}(Z)$ is a factor that accounts for fuel manufacturing tolerances and flux map measurement uncertainty. $F_Q^{MU}(Z)$ is provided in the COLR.

$F_Q^C(Z)$ is an excellent approximation for $F_Q(Z)$ when the reactor is at the steady state power at which the incore flux map was taken.

The expression for $F_Q^W(Z)$ is:

$$F_Q^W(Z) = F_Q^C(Z) * W(Z)$$

where W(Z) is a cycle-dependent function that accounts for power distribution transients encountered during normal operation. W(Z) is included in the COLR. The $F_Q^C(Z)$ is calculated at equilibrium conditions.

The $F_Q(Z)$ limits define limiting values for core power peaking that precludes peak cladding temperatures above 2200°F during either a large or small break LOCA.

This LCO requires operation within the bounds assumed in the safety analyses. Calculations are performed in the core design process to confirm that the core can be controlled in such a manner during operation that it can stay within the LOCA $F_Q(Z)$ limits. If $F_Q(Z)$ cannot be maintained within the LCO limits, reduction of the core power is required and if $F_Q^W(Z)$ cannot be maintained within LCO limits , reduction of the core power is required.

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BASES

Violating the LCO limits for $F_Q(Z)$ may result in an unanalyzed condition while $F_Q(Z)$ is outside its specified limits.

APPLICABILITY

When the OPDMS is inoperable and core power distribution parameters cannot be continuously monitored, it is necessary to determine $F_Q(Z)$ on a periodic basis. Furthermore, the $F_Q(Z)$ limits must be maintained in MODE 1 to prevent core power distributions from exceeding the limits assumed in the safety analyses. Applicability in other MODES is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the reactor coolant to require a limit on the distribution of core power.

ACTIONS

A.1

Reducing THERMAL POWER by ≥ 1% of RTP for each 1% by which $F_Q^C(Z)$ exceeds its limit, maintains an acceptable absolute power density. $F_Q^C(Z)$ is $F_Q^M(Z)$ multiplied by a factor accounting for fuel manufacturing tolerances and flux map measurement uncertainties. $F_Q^M(Z)$ is the measured value of $F_Q(Z)$. The Completion Time of 15 minutes provides an acceptable time to reduce power in an orderly manner without allowing the plant to remain in an unacceptable condition for an extended period of time. The maximum allowable power level initially determined by Required Action A.1 may be affected by subsequent determinations of $F_Q^C(Z)$ and would require power reductions within 15 minutes of the $F_Q^C(Z)$ determination, if necessary to comply with the decreased maximum allowable power level. Decreases in $F_Q^C(Z)$ would allow increasing the maximum allowable power level and increasing power up to this revised limit.

A.2

A reduction of the Power Range Neutron Flux – High Trip setpoints by ≥ 1% for each 1% by which $F_Q^C(Z)$ exceeds its limit is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 72 hours is sufficient considering the small likelihood of a severe transient in this time

ACTIONS (continued)

period and the prompt reduction in THERMAL POWER in accordance

with Required Action A.1. The maximum allowable Power Range

Neutron Flux – High trip setpoints initially determined by Required Action A.2 may be affected by subsequent determinations of $F_Q^C(Z)$ and would require Power Range Neutron Flux – High trip setpoint reductions within 8 hours of the $F_Q^C(Z)$ determination, if necessary to comply with the decreased maximum allowable Power Range Neutron Flux – High trip setpoints. Decreases in $F_Q^C(Z)$ would allow increasing the maximum allowable Power Range Neutron Flux – High trip setpoints.

A.3

Reduction in the Overpower ΔT Trip setpoints (value of K_4) by ≥ 1% for each 1% by which $F_Q^C(Z)$ exceeds its limit is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 72 hours is sufficient considering the small likelihood of a severe transient in this time period and the prompt reduction in THERMAL POWER in accordance with Required Action A.1. The maximum allowable Overpower ΔT trip setpoints initially determined by Required Action A.3 may be affected by subsequent determinations of $F_Q^C(Z)$ and would require Overpower ΔT trip setpoint reductions within 72 hours of the $F_Q^C(Z)$ determination, if necessary to comply with the decreased maximum allowable Overpower ΔT trip setpoints. Decreases in $F_Q^C(Z)$ would allow increasing the maximum allowable Overpower ΔT trip setpoints.

A.4

Verification that $F_Q^C(Z)$ has been restored to within its limit by performing SR 3.2.1.1 and SR 3.2.1.2 prior to increasing THERMAL POWER above the limit imposed by Required Action A.1, assures that core conditions during operation at higher power levels and future operation are consistent with safety analyses assumptions.

Condition A is modified by a Note that requires Required Action A.4 to be performed whenever the Condition is entered. This ensures that SR 3.2.1.1 and SR 3.2.1.2 will be performed prior to increasing THERMAL POWER above the limit of Required Action A.1, even when Condition A is exited prior to performing Required Action A.4. Performance of SR 3.2.1.1 and SR 3.2.1.2 are necessary to assure $F_Q(Z)$ is properly evaluated prior to increasing THERMAL POWER.

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BASES

ACTIONS (continued)

B.1

If it is found that the maximum calculated value of $F_Q(Z)$ which can occur during normal maneuvers, $F_Q^W(Z)$, exceeds its specified limits, there exists a potential for $F_Q^C(Z)$ to become excessively high if a normal operational transient occurs. Reducing the THERMAL POWER by ≥ 1% for each 1% by which $F_Q^W(Z)$ exceeds its limit within the allowed Completion Time of 4 hours maintains an acceptable absolute power density such that even if a transient occurred, core peaking factors would not be exceeded.

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B.2

A reduction of the Power Range Neutron Flux-High trip setpoints by ≥ 1% for each 1% by which the maximum allowable power is reduced, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 72 hours is sufficient considering the small likelihood of a severe transient in this time period and the preceding prompt reduction in THERMAL POWER in accordance with Required Action B.1.

Comment [ss11]: Not applicable to CAOC.

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The implicit assumption is that if W(Z) values were recalculated (consistent with the reduced AFD limits), then $F_Q^C(Z)$ times the recalculated W(Z) values would meet the $F_Q(Z)$ limit. Note that complying with this action (of reducing AFD limits) may also result in a power reduction. Hence the need for B.2, B.3, and B. ... [1]

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B.3

Reduction in the Overpower ΔT trip setpoints value of K_4 by ≥ 1% for each 1% by which the maximum allowable power is reduced, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 72 hours is sufficient considering the small likelihood of a severe transient in this time period, and the preceding prompt reduction in THERMAL POWER in accordance with Required Action B.1.

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ACTIONS (continued)

B.4

Verification that $F_Q^W(Z)$ has been restored to within its limit, by performing SR 3.2.1.1 and SR 3.2.1.2 prior to increasing THERMAL POWER above the maximum allowable power limit imposed by Required Action B.1 ensures that core conditions during operation at higher power levels and future operation are consistent with safety analyses assumptions.

Condition B is modified by a Note that requires Required Action B.4 to be performed whenever the Condition is entered. This ensures that SR 3.2.1.1 and SR 3.2.1.2 will be performed prior to increasing THERMAL POWER above the limit of Required Action B.1, even when

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BASES

Condition A is exited prior to performing Required Action B.4. Performance of SR 3.2.1.1 and SR 3.2.1.2 are necessary to assure $F_Q(Z)$ is properly evaluated prior to increasing THERMAL POWER.

C.1

If Required Actions A.1 through A.4 or B.1 through B.4 are not met within their associated Completion Times, the plant must be placed in a MODE or condition in which the LCO requirements are not applicable. This is done by placing the plant in at least MODE 2 within 6 hours.

This allowed Completion Time is reasonable based on operating experience regarding the amount of time it takes to reach MODE 2 from full power operation in an orderly manner without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1 and SR 3.2.1.2 are modified by two Notes. The first note applies to the situation where the OPDMS is inoperable at the beginning of cycle startup. Note 1 applies during the first power ascension after a refueling. It states that THERMAL POWER may be increased until an equilibrium power level has been achieved at which a power distribution map can be obtained. This allowance is modified, however, by one of the Frequency conditions that requires verification that $F_Q^C(Z)$ and $F_Q^W(Z)$ are within their specified limits after a power rise of more than 10% of RTP over the THERMAL POWER at which they were last verified to be within specified limits. Because $F_Q^C(Z)$ and $F_Q^W(Z)$ could not have previously been measured in this reload core, there is a second Frequency condition, applicable only for reload cores, that requires determination of these parameters before exceeding 75% RTP. This ensures that some determination of $F_Q^C(Z)$ and $F_Q^W(Z)$ are made at a lower power level at

SURVEILLANCE REQUIREMENTS (continued)

which adequate margin is available before going to 100% RTP. Also, this Frequency condition, together with the Frequency condition requiring verification of $F_Q^C(Z)$ and $F_Q^W(Z)$ following a power increase of more than 10%, ensures that they are verified as soon as RTP (or any other level for extended operation) is achieved. In the absence of these Frequency conditions, it is possible to increase power to RTP and operate for 31 days without verification of $F_Q^C(Z)$ and $F_Q^W(Z)$. The Frequency condition is not intended to require verification of these parameters after every 10% increase in power level above the last verification. It only requires verification after a power level is achieved for extended operation that is 10% higher than that power at which $F_Q(Z)$ was last measured.

BASES

The second Note applies to the situation where the OPDMS becomes inoperable while the plant is in MODE 1. Without the continuous monitoring capability of the OPDMS, F_Q limits must be monitored on a periodic basis. The first measurement must be made within 31 days of the most recent date where the OPDMS data has verified peak kw/ft (Z) (and therefore also F_Q) to be within its limit. This is consistent with the 31 day Surveillance Frequency.

SR 3.2.1.1

Verification that $F_Q^C(Z)$ is within its specified limits involves increasing the measured values of $F_Q^C(Z)$ to allow for manufacturing tolerance and measurement uncertainties in order to obtain $F_Q^C(Z)$. Specifically, $F_Q^M(Z)$ is the measured value of $F_Q(Z)$ obtained from incore flux map results and $F_Q^C(Z) = F_Q^M(Z) * F_Q^{MU}(Z)$. $F_Q^C(Z)$ is then compared to its specified limits.

The limit to which $F_Q^C(Z)$ is compared varies inversely with power above 50% RTP.

Performing the Surveillance in MODE 1 prior to exceeding 75% RTP assures that the $F_Q^C(Z)$ limit is met when RTP is achieved because Peaking Factors generally decrease as power level is increased.

SURVEILLANCE REQUIREMENTS (continued)

If THERMAL POWER has been increased by ≥ 10% RTP since the last determination of $F_Q^C(Z)$, another evaluation of this factor is required 12 hours after achieving equilibrium conditions at this higher power level (to assure that $F_Q^C(Z)$ values are being reduced sufficiently with power increase to stay within the LCO limits).

The Frequency of 31 effective full power days (EFPDs) is adequate to monitor the change of power distribution with core burnup because such changes are slow and well controlled when the plant is operated in accordance with Technical Specifications.

SR 3.2.1.2

The nuclear design process includes calculations performed to determine that the core can be operated within the $F_Q(Z)$ limits. Because flux maps are taken in steady state conditions, the variations in power distribution resulting from normal operational maneuvers are not present in the flux map data. These variations are, however, conservatively calculated by considering a wide range of unit maneuvers in normal operation. The maximum peaking factor increase over steady state values, calculated as

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BASES

a function of core elevation, Z, is called W(Z). Multiplying the measured total peaking factor, $F_Q^C(Z)$, by W(Z) gives the maximum $F_Q(Z)$ calculated to occur in normal operation, $F_Q^W(Z)$.

The limit to which $F_Q^W(Z)$ is compared varies inversely with power.

The W(Z) curve is provided in the COLR for discrete core elevations. $F_Q^W(Z)$ evaluations are not applicable for the following axial core regions, measured in percent of core height:

a. Lower core region, from 0% to 8% inclusive; and

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b. Upper core region, from 92% to 100% inclusive.

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The top and bottom 8% of the core are excluded from the evaluation because of the difficulty of making a precise measurement in these regions and because of the low probability that these regions would be more limiting than the safety analyses.

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This Surveillance has been modified by a Note, which may require that more frequent surveillances be performed. If $F_Q^W(Z)$ is evaluated and found to be within its limit, an evaluation of the expression below is

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SURVEILLANCE REQUIREMENTS (continued)

required to account for any increase to $F_Q^M(Z)$ which could occur and cause the $F_Q(Z)$ limit to be exceeded before the next required $F_Q(Z)$ evaluation.

If the two most recent FQ(Z) evaluations show an increase in $F_Q^C(Z)$, it is required to meet the $F_Q(Z)$ limit with the last $F_Q^W(Z)$ increased by the greater of a factor of 1.02 or by an appropriate factor as specified in the COLR or to evaluate $F_Q(Z)$ more frequently, each 7 EFPDs. These alternative requirements will prevent $F_Q(Z)$ from exceeding its limit for any significant period of time without detection.

Performing the Surveillance in MODE 1 prior to exceeding 75% of RTP ensures that the $F_Q(Z)$ limit will be met when RTP is achieved, because peaking factors are generally decreased as power level is increased.

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The Surveillance Frequency of 31 EFPDs is adequate to monitor the change of power distribution because such a change is sufficiently slow, when the plant is operated in accordance with Technical Specifications, to preclude the occurrence of adverse peaking factors between 31 EFPD Surveillances. The Surveillance may be done more frequently if required by the results of $F_Q(Z)$ evaluations.

$F_Q(Z)$ is verified at power increases of at least 10% RTP above the THERMAL POWER of its last verification, 12 hours after achieving equilibrium conditions, to assure that $F_Q(Z)$ will be within its limit at higher power levels.

REFERENCES

1. 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors," 1974.
2. NUREG-0800, Standard Review Plan (SRP), Section 4.2, Revision 3, "Fuel System Design," (Appendix B)
3. 10 CFR 50, Appendix A, GDC 26.
4. WCAP-7308-L-P-A, "Evaluation of Nuclear Hot Channel Factor Uncertainties," June 1988 (Westinghouse Proprietary) and WCAP-7308-L-A (Non-Proprietary).
5. WCAP-10216-P-A, Revision 1A, "Relaxation of Constant Axial Offset Control FQ Surveillance Technical Specification," February 1994 (Westinghouse Proprietary) and WCAP-10217-A (Non-Proprietary).

Deleted: Regulatory Guide 1.77, Rev. 0, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors," May 1974.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2 Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)

BASES

BACKGROUND

The purpose of this LCO is to establish limits on the power density at any point in the core so that the fuel design criteria are not exceeded and the accident analysis assumptions remain valid. The design limits on local (pellet) and integrated fuel rod peak power density are expressed in terms of hot channel factors. Control of the core power distribution with respect to these factors assures that local conditions in the fuel rods and coolant channels do not challenge core integrity at any location during either normal operation or a postulated accident analyzed in the safety analyses.

$F^N_{\Delta H}$ is defined as the ratio of the integral of the linear power along the fuel rod with the highest integrated power to the average integrated fuel rod power. Therefore, $F^N_{\Delta H}$ is a measure of the maximum total power produced in a fuel rod.

$F^N_{\Delta H}$ is sensitive to fuel loading patterns, bank insertion and fuel burnup. $F^N_{\Delta H}$ typically increases with control bank insertion and typically decreases with fuel burnup.

With the On-line Power Distribution Monitoring System (OPDMS) OPERABLE, $F^N_{\Delta H}$ is determined continuously by the OPDMS. When the OPDMS is inoperable, $F^N_{\Delta H}$ is not directly measurable but is inferred from a power distribution map obtained with the incore detector system. Specifically, the results of the three dimensional power distribution map are analyzed to determine $F^N_{\Delta H}$. This factor is calculated at least every 31 effective full power days (EFPDs). Also, during power operation with the OPDMS inoperable, the global power distribution is monitored by LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," which are directly and continuously measured process variables.

The COLR provides peaking factor limits that ensure that the design basis value of the departure from nucleate boiling (DNB) is met for normal operation, operational transients, and any transient condition arising from events of moderate frequency. The DNB design basis precludes DNB and is met by limiting the minimum local DNB heat flux ratio. Transient

BACKGROUND (continued)

events that may be DNB limited are assumed to begin with a $F^N_{\Delta H}$ that satisfies the LCO requirements.

Operation outside the LCO limits may produce unacceptable consequences if a DNB limiting event occurs. The DNB design basis ensures that there is no overheating of the fuel that results in possible cladding perforation with the release of fission products to the reactor coolant.

APPLICABLE SAFETY ANALYSES

Limits on $F^N_{\Delta H}$ prevent core power distributions from occurring which would exceed the following fuel design limits:

a. There must be at least a 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the hottest fuel rod in the core does not experience a DNB condition;

b. During a large break loss of coolant accident (LOCA), the peak cladding temperature (PCT) must not exceed 2200°F;

c. During an ejected rod accident, the energy deposition to the fuel must not exceed applicable limits (Ref. 1); and

d. Fuel design limits required by GDC 26 (Ref. 2) for the condition when the control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn.

For transients that may be DNB limited, the Reactor Coolant System (RCS) flow and $F^N_{\Delta H}$ are the core parameters of most importance. The limits on $F^N_{\Delta H}$ ensure that the DNB design basis is met for normal operation, operational transients, and any transients arising from events of moderate frequency. The DNB design basis is met by limiting the minimum DNB ratio (DNBR) to the 95/95 DNB criterion. This value provides a high degree of assurance that the hottest fuel rod in the core will not experience a DNB.

The allowable $F^N_{\Delta H}$ limit increases with decreasing power level. This functionality in $F^N_{\Delta H}$ is included in the analyses that provide the Reactor Core Safety Limits (SLs) of SL 2.1.1. Therefore, any DNB events in which the calculation of the core limits is modeled implicitly use this

APPLICABLE SAFETY ANALYSES (continued)

variable value of $F^N_{\Delta H}$ in the analyses. Likewise, all transients that may be DNB limited are assumed to begin with an initial $F^N_{\Delta H}$ as a function of power level defined by the COLR limit equation.

The LOCA safety analysis indirectly models $F^N_{\Delta H}$ as an input parameter. The Nuclear Heat Flux Hot Channel Factor ($F_Q(Z)$) and the axial peaking factors are inserted directly into the LOCA safety analyses that verify the acceptability of the resulting peak cladding temperature (Ref. 3).

The fuel is protected in part by Technical Specifications, which provide assurance that the initial conditions assumed in the safety and accident analyses remain valid. With the OPDMS OPERABLE, peak kw/ft(Z) and $F^N_{\Delta H}$ are directly monitored. Should the OPDMS become inoperable, the following LCOs assure that the conditions assumed for the safety analysis remain valid: LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," LCO 3.1.6, "Control Bank Insertion Limits," LCO 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)," and LCO 3.2.1, "Heat Flux Hot Channel Factor ($F_Q(Z)$)."

When the OPDMS is not available to measure power distribution parameters continuously, $F^N_{\Delta H}$ and $F_Q(Z)$ are measured periodically using the incore detector system. Measurements are generally taken with the core at, or near, steady-state conditions. Without the OPDMS, core monitoring and control under transient conditions (Condition 1 events) are accomplished by operating the core within the limits of the LCOs on AFD, QPTR, and Bank Insertion Limits.

$F^N_{\Delta H}$ satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

$F^N_{\Delta H}$ shall be maintained within the limits of the relationship provided in the COLR.

The $F^N_{\Delta H}$ limit identifies the coolant flow channel with the maximum enthalpy rise. This channel has the least heat removal capability and thus the highest probability for a DNB.

LCO (continued)

The limiting value of $F^N_{\Delta H}$, described by the equation contained in the COLR, is the design radial peaking factor used in the unit safety analyses.

A power multiplication factor in this equation includes an additional margin for higher radial peaking from reduced thermal feedback and greater control rod insertion at low power levels. The limiting value of $F^N_{\Delta H}$ is allowed to increase 0.3% for every 1% RTP reduction in THERMAL POWER.

APPLICABILITY

When the OPDMS is inoperable and core power distribution parameters cannot be continuously monitored, it is necessary to monitor $F^N_{\Delta H}(Z)$ on a periodic basis. Furthermore, $F^N_{\Delta H}$ limits must be maintained in MODE 1 to preclude core power distributions from exceeding the fuel design limits for DNBR and peak cladding temperature (PCT). Applicability in other modes is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the coolant to require a limit on the distribution of core power. Specifically, the design bases events that are sensitive to $F^N_{\Delta H}$ in other modes (MODES 2 through 5) have significant margin to DNB, and therefore, there is no need to restrict $F^N_{\Delta H}$ in these modes.

ACTIONS

A.1.1

With $F^N_{\Delta H}$ exceeding its limit, the unit is allowed 4 hours to restore $F^N_{\Delta H}$ to within its limits. This restoration may, for example, involve realigning any misaligned rods or reducing power enough to bring $F^N_{\Delta H}$ within its power-dependent limit.

When the $F^N_{\Delta H}$ limit is exceeded, it is not likely that the DNBR limit would be violated in steady state operation, since events that could significantly perturb the $F^N_{\Delta H}$ value (e.g., static control rod misalignment) are considered in the safety analyses. However, the DNBR limit may be violated if a DNB limiting event occurs. Thus, the allowed Completion Time of 4 hours provides an acceptable time to restore $F^N_{\Delta H}$ to within its limits without allowing the plant to remain outside $F^N_{\Delta H}$ limits for an extended period of time.

ACTIONS (continued)

Condition A is modified by a Note that requires that Required Actions A.2 and A.3 must be completed whenever Condition A is entered. Thus, if power is not reduced because this Required Action is completed within the 4 hour time period, Required Action A.2 would nevertheless require another measurement and calculation of $F^N_{\Delta H}$ within 24 hours in accordance with SR 3.2.2.1.

However, if power were reduced below 50% RTP, Required Action A.3 requires that another determination of $F^N_{\Delta H}$ must be done prior to exceeding 50% RTP, prior to exceeding 75% RTP, and within 24 hours after reaching or exceeding 95% RTP. In addition, Required Action A.2 would be performed if power ascension were delayed past 24 hours.

A.1.2.1 and A.1.2.2

If the value of $F^N_{\Delta H}$ is not restored to within its specified limit either by adjusting a misaligned rod or by reducing THERMAL POWER, the alternative option is to reduce THERMAL POWER to < 50% RTP in accordance with Required Action A.1.2.1 and reduce the Power Range Neutron Flux - High to ≤ 55% RTP in accordance with Required Action A.1.2.2. The reduction in trip setpoints ensures that continuing operation remains at an acceptable low power level with adequate DNBR margin. The allowed Completion Time of 4 hours for Required Action A.1.2.1 is consistent with those specified in Required Action A.1.1 and provides an acceptable time to reach the required power level from full power operation without allowing the plant to remain in an unacceptable condition for an extended period of time. The Completion Time of 4 hours for Required Actions A.1.1 and A.1.2.1 are not additive.

The allowed Completion Time of 72 hours to reset the trip setpoints per Required Action A.1.2.2 recognizes that, once power is reduced, the safety analysis assumptions are satisfied and there is no urgent need to reduce the trip setpoints. This is a sensitive operation that may cause an inadvertent reactor trip.

A.2

Once the power level has been reduced to < 50% RTP per Required Action A.1.2.1, an incore flux map (SR 3.2.2.1) must be obtained and the measured value of $F^N_{\Delta H}$ verified not to exceed the allowed limit at the lower power level. The unit is provided 20 additional hours to perform this task over and above the 4 hours allowed by either Action A.1.1 or Action A.1.2.1. The Completion Time of 24 hours is acceptable because

ACTIONS (continued)

of the increase in the DNB margin, which is obtained at lower power levels, and the low probability of having a DNB limiting event within this 24 hour period. Additionally, operating experience has indicated that this Completion Time is sufficient to obtain the incore flux map, perform the required calculations, and evaluate $F^N_{\Delta H}$.

A.3

Verification that $F^N_{\Delta H}$ is within its specified limits after an out of limit occurrence assures that the cause that led to the $F^N_{\Delta H}$ exceeding its limit is corrected, and that subsequent operation will proceed within the LCO limit. This Action demonstrates that the $F^N_{\Delta H}$ limit is within the LCO limits prior to exceeding 50% of RTP, again prior to exceeding 75% RTP, and within 24 hours after THERMAL POWER is ≥ 95% RTP.

This Required Action is modified by a Note, that states that THERMAL POWER does not have to be reduced prior to performing this action.

B.1

When Required Actions A.1.1 through A.3 cannot be completed within their required Completion Times, the plant must be placed in a mode in which the LCO requirements are not applicable. This is done by placing the plant in at least MODE 2 within 6 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience regarding the time required to reach MODE 2 from full power conditions in an orderly manner without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1

When the OPDMS is OPERABLE, the value of $F^N_{\Delta H}$ is directly and continuously monitored. With the OPDMS inoperable, the value of $F^N_{\Delta H}$ is determined by using the incore detector system to obtain a flux distribution map. A data reduction computer program then calculates the maximum value of $F^N_{\Delta H}$ from the measured flux distributions. The measured value of $F^N_{\Delta H}$ must be multiplied by a measurement uncertainty factor before making comparisons to the $F^N_{\Delta H}$ limit.

SURVEILLANCE REQUIREMENTS (continued)

After each refueling, with the OPDMS inoperable, $F^N_{\Delta H}$ must be determined prior to exceeding 75% RTP. This requirement ensures that $F^N_{\Delta H}$ limits are met at the beginning of each fuel cycle.

With the OPDMS inoperable, the 31 EFPDs Frequency is acceptable because the power distribution will change relatively slowly over this amount of fuel burnup. This Frequency is short enough so that the $F^N_{\Delta H}$ limit will not be exceeded for any significant period of operation.

REFERENCES

1. NUREG-0800, Standard Review Plan (SRP), Section 4.2, Revision 3, "Fuel System Design," (Appendix B).

2. 10 CFR 50, Appendix A, GDC 26.

3. 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors."

Deleted: Regulatory Guide 1.77, Rev. 0, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors," May 1974.

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B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3 AXIAL FLUX DIFFERENCE (AFD) (Constant Axial Offset Control (CAOC) Methodology)

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BASES

BACKGROUND

The purpose of this LCO is to establish limits on the values of the AFD in order to limit the amount of axial power distribution skewing to either the top or bottom of the core when the On-Line Power Distribution Monitoring System (OPDMS) is inoperable. By limiting the amount of power distribution skewing, core peaking factors are consistent with the assumptions used in the safety analyses. Limiting power distribution skewing over time also minimizes the xenon distribution skewing which is a significant factor in axial power distribution control.

The operating strategy used to control the axial power distribution, Constant Axial Offset Control (CAOC), involves maintaining the AFD within a tolerance band about a burnup dependent target, known as the target flux difference, to minimize the variation of the axial peaking factor and axial xenon distribution during unit maneuvers.

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Deleted: is a calculational procedure which defines the allowed operational space of the AFD versus THERMAL POWER. The AFD limits are

Deleted: selected by considering a range of axial xenon distributions that may occur as a result of large variations of the AFD. Subsequently, power peaking factors and power distributions are examined to assure that the loss of coolant accident (LOCA), loss of flow accident, and anticipated transient limits are met. Violation of the AFD limits invalidate the conclusions of the accident and transient analyses with regard to fuel cladding integrity.

The target flux difference is determined with reactor power at or near the RTP value, equilibrium xenon conditions, and the control banks positioned at or near a nominally assumed position for steady state operation at high power levels. The value of the target flux difference obtained at these conditions divided by the fraction of RTP is the target flux difference at RTP for the associated core burnup conditions. Target flux differences for other THERMAL POWER levels are obtained by multiplying the RTP value by the appropriate fractional THERMAL POWER level.

The AFD is monitored on an automatic basis using the computer which has an AFD monitor alarm. The frequency of monitoring the AFD by the computer is frequent enough (typically once per minute) to provide an essentially continuous accumulation of penalty deviation time that allows the operator to assess the status of the penalty deviation time. The computer determines a time average (typically over 1 minute) of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for two or more OPERABLE excore channels is outside the target band and the THERMAL POWER is > 90% RTP. During operation at THERMAL POWER levels < 90% RTP but > 15% RTP, the computer sends an alarm message when the cumulative penalty deviation time is > 1 hour in the previous 24 hours.

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Periodic updating of the target flux difference value is necessary to follow

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the change of the flux difference at steady state conditions with burnup.

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The Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$) and QPTR LCOs limit the radial component of the peaking factors.

Deleted: The CAOC operating space is typically smaller and lies within the RAOC operating space. Control within the CAOC operating space constrains the variation of axial xenon distributions and axial power distributions. RAOC calculations assume a wide range of xenon distributions and then confirm that the resulting power distributions satisfy the requirements of the accident analyses.

It is expected that AFD will normally be automatically controlled by the Plant Control System (PLS), which will demand movement of the AO bank to maintain AFD within a control dead-band around the target flux difference. The control dead-band is defined to be more restrictive that the CAOC limits associated with this LCO such that automatic control can typically be expected to maintain adherence to this LCO.

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BASES

APPLICABLE SAFETY ANALYSES

The AFD is a measure of the axial power distribution skewing to either the top or bottom half of the core. The AFD is sensitive to many core related parameters such as control bank positions, core power level, axial burnup, axial xenon distribution, and, to a lesser extent, reactor coolant temperature and boron concentration.

The allowed range of the AFD is used in the nuclear design process to confirm that operation within these limits produces core peaking factors and axial power distributions that meet safety analysis requirements.

The CAOC methodology (Refs. 1, 2 and 3) entails:

a. Establishing an envelope of allowed power shapes and power densities,

b. Devising an operating strategy for the cycle that maximizes unit flexibility (maneuvering) and minimizes axial power shape changes,

c. Demonstrating that this strategy does not result in core conditions that violate the envelope of permissible core power characteristics, and

d. Demonstrating that this power distribution control scheme can be effectively supervised with excore detectors.

Deleted: Three dimensional power distribution calculations are performed to demonstrate that normal operation power shapes are acceptable for the LOCA, the loss of flow accident, and for initial conditions of anticipated transients (Ref. 2). The tentative limits are adjusted as necessary to meet the safety analysis requirements.

With the OPDMS inoperable, the limits on the AFD ensure that the Heat Flux Hot Channel Factor ($F_Q(Z)$) is not exceeded during either normal operation or in the event of xenon redistribution following power changes. The limits on the AFD also restrict the range of power distributions that are used as initial conditions in the analyses of Condition 2, 3, or 4 events. This ensures that the fuel cladding integrity is maintained for these postulated accidents. The most important Condition 4 event is the LOCA. The most important Condition 3 event is the loss of flow accident.

Deleted: RAOC

BASES

The most important Condition 2 events are uncontrolled bank withdrawal and boration or dilution accidents. Condition 2 accidents simulated to begin from within the AFD limits are used to confirm the adequacy of the Overpower ΔT and Overtemperature ΔT trip setpoints.

The limits on the AFD satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The shape of the power profile in the axial (i.e., the vertical) direction is largely under the control of the operator through the manual operation of the control banks or automatic motion of control banks. The automatic motion of the control banks is in response to temperature deviations resulting from manual operation of the Chemical and Volume Control System (CVS) to change boron concentration or from power level changes.

Signals are available to the operator from the Protection and Safety Monitoring System (PMS) excore neutron detectors (Ref. 4). Separate signals are taken from the top and bottom detectors. The AFD is defined as the difference in normalized flux signals between the top and bottom

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LCO (continued)

excore detectors in each detector well. For convenience, this flux difference is converted to provide flux difference units expressed as a percentage and labeled as %Δ flux or %ΔI.

The required target band varies with axial burnup distribution, which in turn varies with the core average accumulated burnup. The target band defined in the COLR may provide one target band for the entire cycle or more than one band, each to be followed for a specific range of cycle burnup. With THERMAL POWER ≥ 90% RTP, the AFD must be kept within the target band. With the AFD outside the target band with THERMAL POWER ≥ 90% RTP, the assumptions of the accident analyses may be violated.

Deleted: The AFD limits are provided in the COLR. Figure B 3.2.3-1 shows typical RAOC AFD limits. The AFD limits for RAOC do not depend on the target flux difference. However, the target flux difference may be used to minimize changes in the axial power distribution.

Violating this LCO on the AFD, with the OPDMS inoperable, could produce unacceptable consequences if a Condition 2, 3 or 4 event occurs while the AFD is outside its specified limits.

Figure B 3.2.3-1 shows a typical target band and typical AFD acceptable operation limits.

The LCO is modified by four Notes. Note 1 states the conditions necessary for declaring the AFD outside of the target band. Notes 2 and 3 describe how the cumulative penalty deviation time is calculated. It is intended that the unit is operated with the AFD within the target band about the target flux difference. However, during rapid THERMAL POWER reductions, control bank motion may cause the AFD to deviate

BASES

outside of the target band at reduced THERMAL POWER levels. This deviation does not affect the xenon distribution sufficiently to change the envelope of peaking factors that may be reached on a subsequent return to RTP with the AFD within the target band, provided the time duration of the deviation is limited. Accordingly, while THERMAL POWER is ≥ 50% RTP and < 90% RTP (i.e., Part b of this LCO), a 1 hour cumulative penalty deviation time limit, cumulative during the preceding 24 hours, is allowed during which the unit may be operated outside of the target band but within the acceptable operation limits provided in the COLR (Note 2). This penalty time is accumulated at the rate of 1 minute for each 1 minute of operating time within the power range of Part b of this LCO (i.e., THERMAL POWER ≥ 50% RTP). The cumulative penalty time is the sum of penalty times from Parts b and c of this LCO.

For THERMAL POWER levels > 15% RTP and < 50% RTP (i.e., Part c of this LCO), deviations of the AFD outside of the target are less significant. Note 3 allows the accumulation of ½ minute penalty deviation time per 1 minute of actual time outside of the target band and reflects this reduced significance. With THERMAL POWER < 15% RTP, AFD is not a significant parameter in the assumptions used in the safety analysis and therefore requires no limits. Because the xenon distribution produced at THERMAL POWER levels less than RTP does affect the power distribution as power is increased, unanalyzed xenon and power distribution is prevented by limiting the accumulated penalty deviation time.

For surveillance of the power range channels performed according to SR 3.3.1.5, Note 4 allows deviation outside the target band for 16 hours and no penalty deviation time accumulated. Some deviation in the AFD is required for doing the NIS calibration with the incore detector system. This calibration is performed every 92 days.

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APPLICABILITY

The AFD requirements are applicable in MODE 1 above 15% RTP and OPDMS inoperable. Above 50% RTP, the combination of THERMAL POWER and core peaking factors are of primary importance in safety analysis (Ref. 1).

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Between 15% RTP and 90% RTP, this LCO is applicable to ensure that the distributions of xenon are consistent with safety analysis assumptions.

At or below 15% RTP and for lower operating MODES, the stored energy in the fuel and the energy being transferred to the reactor coolant are low. The value of the AFD in these conditions does not affect the consequences of the design basis events.

Low signal levels in the excore channels may preclude obtaining valid

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BASES

AFD signals below 15% RTP.

Deleted: For AFD limits developed using RAOC methodology, the value of the AFD does not affect the limiting accident consequences with THERMAL POWER < 50% RTP and for lower operating power MODES. With the OPDMS inoperable, it is necessary to monitor AFD via the excore detectors to ensure that it remains within the RAOC limits.

ACTIONS

A.1

With the AFD outside the target band and THERMAL POWER ≥ 90% RTP, the assumptions used in the accident analyses may be violated with respect to the maximum heat generation. Therefore, a Completion Time of 15 minutes is allowed to restore the AFD to within the target band because xenon distributions change little in this relatively short time.

B.1

If the AFD cannot be restored within the target band, then reducing THERMAL POWER to < 90% RTP places the core in a condition that has been analyzed and found to be acceptable, provided that the AFD is within the acceptable operation limits provided in the COLR.

The allowed Completion Time of 15 minutes provides an acceptable time to reduce power to < 90% RTP without allowing the plant to remain in an unanalyzed condition for an extended period of time.

C.1

With THERMAL POWER < 90% RTP but ≥ 50% RTP, operation with the AFD outside the target band is allowed for up to 1 hour if the AFD is within the acceptable operation limits provided in the COLR. With the AFD within these limits, the resulting axial power distribution is acceptable as an initial condition for accident analyses assuming the then existing xenon distributions. The 1 hour cumulative penalty deviation time restricts the extent of xenon redistribution. Without this limitation, unanalyzed xenon axial distributions may result from a different pattern of xenon buildup and decay. The reduction to a power level < 50% RTP puts the reactor at a THERMAL POWER level at which the AFD is not a significant accident analysis parameter.

If the indicated AFD is outside the target band and outside the acceptable operation limits provided in the COLR, the peaking factors assumed in accident analysis may be exceeded with the existing xenon condition. (Any AFD within the target band is acceptable regardless of its relationship to the acceptable operation limits.) The Completion Time of 30 minutes allows for a prompt, yet orderly, reduction in power.

Condition C is modified by a Note that requires that Required Action C.1 must be completed whenever this Condition is entered.

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BASES

D.1

If Required Action C.1 is not completed within its required Completion Time of 30 minutes, the axial xenon distribution starts to become significantly skewed with the THERMAL POWER ≥ 50% RTP. In this situation, the assumption that a cumulative penalty deviation time of 1 hour or less during the previous 24 hours while the AFD is outside its target band is acceptable at < 50% RTP, is no longer valid.

Reducing the power level to < 15% RTP within the Completion Time of 9 hours and complying with LCO penalty deviation time requirements for subsequent increases in THERMAL POWER ensure that acceptable xenon conditions are restored.

This Required Action must also be implemented either if the cumulative penalty deviation time is > 1 hour during the previous 24 hours, or the AFD is not within the target band and not within the acceptable operation limits.

Deleted: Required Action A.1 requires a THERMAL POWER reduction to < 50% RTP. This places the core in a condition where the value of the AFD is not important in the applicable safety analyses. A Completion Time of 30 minutes is reasonable, based on operating experience, to reach 50% RTP without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1

This surveillance verifies that the AFD, as indicated by the PMS excore channel, is within the target band. The Surveillance Frequency of 7 days is adequate considering that the AFD is monitored by a computer and any deviation from requirements is alarmed.

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The AFD should be monitored and logged more frequently in periods of operation for which the power level or control bank positions are changing to allow corrective measures when the AFD is more likely to move outside the target band.

SR 3.2.3.2

This Surveillance requires that the target flux difference is updated at a Frequency of 31 effective full power days (EFPD) to account for small changes that may occur in the target flux differences in that period due to burnup by performing SR 3.2.3.3.

Alternatively, linear interpolation between the most recent measurement of the target flux differences and a predicted end of cycle value provides a reasonable update because the AFD changes due to burnup tend toward 0% AFD. When the predicted end of cycle AFD from the cycle nuclear design is different from 0%, it may be a better value for the interpolation.

SR 3.2.3.3

Deleted: RAOC

BASES

Measurement of the target flux difference is accomplished by taking a flux map when the core is at equilibrium xenon conditions, preferably at high power levels with the control banks at a reference position. This flux map provides the equilibrium xenon axial power distribution from which the target value can be determined. The target flux difference varies slowly with core burnup.

A Frequency of 31 EFPD after each refueling and 92 EFPD thereafter for remeasuring the target flux differences adjusts the target flux difference for each excore channel to the value measured at steady state conditions. This is the basis for the CAOC. Remeasurement at this Surveillance interval also establishes the AFD target flux difference values that account for changes in incore excore calibrations that may have occurred in the interim.

A Note modifies this SR to allow the predicted beginning of cycle AFD from the cycle nuclear design to be used to determine the initial target flux difference after each refueling.

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REFERENCES

1. WCAP-8385, "Power Distribution Control and Load Following Procedures," Westinghouse Electric Corporation, September 1974 (Westinghouse Proprietary) and WCAP-8403 (Non-Proprietary).

Deleted: 2. R. W. Miller et al., "Relaxation of Constant Axial Offset Control: F_Q Surveillance Technical Specification," WCAP-10216-P-A, June 1983 (Westinghouse Proprietary) and WCAP-10217-A (Non-Proprietary).

2. T. M. Anderson to K. Kniel (Chief of Core Performance Branch, NRC), Attachment: "Operation and Safety Analysis Aspects of an Improved Load Follow Package," January 31, 1980.

3. C. Eicheldinger to D. B. Vassallo (Chief of Light Water Reactors Branch, NRC), Letter NS-CE-687, July 16, 1975.

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4. Chapter 15, "Accident Analysis."

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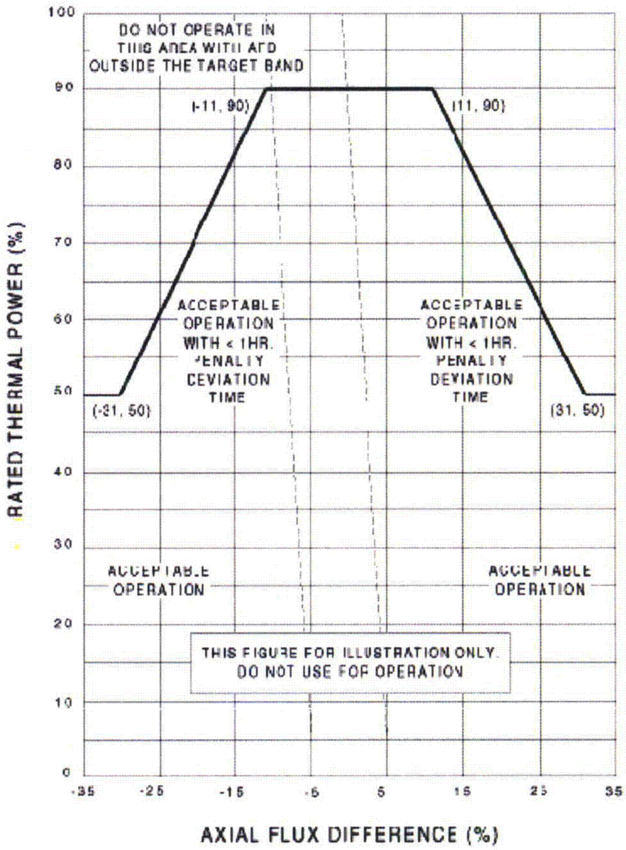

THIS FIGURE IS FOR ILLUSTRATION ONLY. DO NOT USE FOR OPERATION.

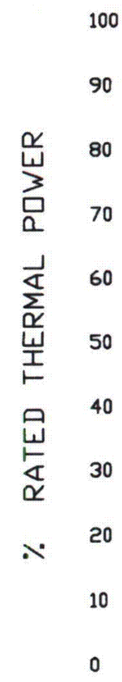

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Figure B 3.2.3-1 (page 1 of 1)
AXIAL FLUX DIFFERENCE Acceptable Operation Limits and Target Band Limits as a Function of RATED THERMAL POWER

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 QUADRANT POWER TILT RATIO (QPTR)

BASES

BACKGROUND

With the Online Power and Distribution Monitoring System (OPDMS) inoperable, the QPTR limit ensures that the gross radial power distribution remains consistent with the design values used in the safety analyses. Precise radial power distribution measurements are made during startup testing, after refueling, and periodically during power operation.

The power density at any point in the core must be limited so that the fuel design criteria are maintained. With the OPDMS OPERABLE, the peak kw/ft(Z) is continuously and directly monitored. With the OPDMS inoperable, LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," and LCO 3.1.6, "Control Rod Insertion Limits," provide limits on process variables that characterize and control the three dimensional power distribution of the reactor core. Control of these variables ensures that the core operates within the fuel design criteria and that the power distribution remains within the bounds used in the safety analyses.

APPLICABLE SAFETY ANALYSES

This LCO precludes core power distributions from occurring which would violate the following fuel design criteria:

a. During a large break loss of coolant accident (LOCA), the peak cladding temperature (PCT) must not exceed 2200°F (Ref. 1);

b. During a loss of forced reactor coolant flow accident, there must be at least a 95% probability at a 95% confidence level (the 95/95 departure from nucleate boiling (DNB) criterion) that the hot fuel rod in the core does not experience a DNB condition;

c. During an ejected rod accident, the energy deposition to the fuel must not exceed applicable limits (Ref. 2); and

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d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 3).

The LCO limits on the AFD, the QPTR, the Heat Flux Hot Channel Factor ($F_Q(Z)$), the Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$), and control bank insertion are established to preclude core power distributions from occurring which would exceed the safety analyses limits.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Should the OPDMS become inoperable, the QPTR limits ensure that $F_{\Delta H}^{N}$ and $F_Q(Z)$ remain below their limiting values by preventing an undetected change in the gross radial power distribution.

In MODE 1, with the OPDMS inoperable, the $F_{\Delta H}^{N}$ and $F_Q(Z)$ limits must be maintained to preclude core power distributions from exceeding design limits assumed in the safety analyses.

The QPTR satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The QPTR limit of 1.02, where corrective action is required, provides a margin of protection for both the DNB ratio (DNBR) and linear heat generation rate contributing to excessive power peaks resulting from X-Y plane power tilts. A limiting QPTR of 1.02 can be tolerated before the margin for uncertainty in $F_Q(Z)$ and $F_{\Delta H}^{N}$ is possibly challenged.

APPLICABILITY

The QPTR limit must be maintained in MODE 1 with THERMAL POWER > 50% RTP to preclude core power distributions from exceeding the design limits. With the OPDMS inoperable, a continuous on-line indication of core peaking factors is not available. Therefore, QPTR must be monitored and the limits on QPTR ensure that peaking factors will be within design limits.

Applicability in MODE 1 ≤ 50% RTP and in other MODES is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the reactor coolant to require the implementation of a QPTR limit on the distribution of core power. The QPTR limit in these conditions is, therefore, not important. Note that the $F_{\Delta H}^{N}$ and $F_Q(Z)$ LCOs still apply, but allow progressively higher peaking factors at 50% RTP or lower.

ACTIONS

A.1

With the QPTR exceeding its limit, and the OPDMS inoperable, a power level reduction of 3% RTP for each 1% by which the QPTR exceeds 1.00 is a conservative tradeoff of total core power with peak linear power. The Completion Time of 2 hours allows sufficient time to identify the cause and correct the tilt. Note that the power reduction itself may cause a change in the tilted condition.

ACTIONS (continued)

The maximum allowable power level initially determined by Required Action A.1 may be affected by subsequent determinations of QPTR. Increases in QPTR would require power reduction within 2 hours of QPTR determination, if necessary to comply with the decreased maximum allowable power level and increasing power up to this revised limit.

A.2

After completion of Required Action A.1, the QPTR alarm may be in its alarmed state. As such, any additional changes in the QPTR are detected by requiring a check of the QPTR once per 12 hours thereafter. A 12 hour Completion Time is sufficient because any additional change in QPTR would be relatively slow.

A.3

The peaking factors $F_Q(Z)$, as approximated by $F_Q^C(Z)$ and $F_Q^W(Z)$, and $F_{\Delta H}^N$ are of primary importance in assuring that the power distribution remains consistent with the initial conditions used in the safety analyses. Performing SRs on $F_{\Delta H}^N$ and $F_Q(Z)$ within the Completion Time of 24 hours after achieving equilibrium conditions from a Thermal Power reduction power Required Action A.1 ensures that these primary indicators of power distribution are within their respective limits. A Completion Time of 24 hours after achieving equilibrium conditions from a Thermal Power reduction power Required Action A.1 takes into consideration the rate at which peaking factors are likely to change, and the time required to stabilize the plant and perform a flux map. If these peaking factors are not within their limits, the Required Actions of these Surveillances provide an appropriate response for the abnormal condition. If the QPTR remains above its specified limits, the peaking factor surveillances are required each 7 days thereafter to evaluate $F_{\Delta H}^N$ and $F_Q(Z)$ with changes in power distribution. Relatively small changes are expected due to either burnup and xenon redistribution or correction of the cause for exceeding the QPTR limit.

A.4

Although $F_{\Delta H}^N$ and $F_Q(Z)$ are of primary importance as initial conditions in the safety analyses, other changes in the power distribution may occur as the QPTR limit is exceeded and may have an impact on the validity of the safety analysis. A change in the power distribution can affect such

ACTIONS (continued)

reactor parameters as bank worths and peaking factors for rod malfunction accidents. When the QPTR exceeds its limit, it does not necessarily mean a safety concern exists. It does mean that there is an indication of a change in the gross radial power distribution that requires an investigation and evaluation that is accomplished by examining the incore power distribution. Specifically, the core peaking factors and the quadrant tilt must be evaluated because they are the factors which best characterize the core power distribution. This re-evaluation is required to assure that, before increasing THERMAL POWER to above the limit of Required Action A.1, the reactor core conditions are consistent with the assumptions in the safety analyses.

A.5

If the QPTR has exceeded the 1.02 limit and a re-evaluation of the safety analysis is completed and shows that safety requirements are met, the excore detectors are normalized to restore QPTR to within limits prior to increasing THERMAL POWER to above the limit of Required Action A.1. Normalization is accomplished in such a manner that the indicated QPTR following normalization is near 1.00. This is done to detect any subsequent significant changes in QPTR.

Required Action A.5 is modified by two Notes. Note 1 states that the QPTR is not restored to within limits until after the re-evaluation of the safety analysis has determined that core conditions at RTP are within the safety analysis assumptions (i.e., Required Action A.4). Note 2 states that if Required Action A.5 is performed, then Required Action A.6 shall be performed. Required Action A.5 normalizes the excore detectors to restore QPTR to within limits, which restores compliance with LCO 3.2.4. Thus, Note 2 prevents exiting the Actions prior to completing flux mapping to verify peaking factors, per Required Action A.6. These Notes are intended to prevent any ambiguity about the required sequence of actions.

A.6

Once the flux tilt is restored to within limits (i.e., Required Action A.5 is performed), it is acceptable to return to full power operation. However, as an added check that the core power distribution is consistent with the safety analysis assumptions, Required Action A.6 requires verification that $F_Q(Z)$ as approximated by $F_Q^C(Z)$ and $F_Q^W(Z)$, and $F_{\Delta H}^N$ are within their specified limits within 24 hours of achieving equilibrium conditions at RTP. As an added precaution, if the core power does not reach equilibrium conditions at RTP within 24 hours, but is increased slowly,

BASES

ACTIONS (continued)

then the peaking factor surveillances must be performed within 48 hours after increasing THERMAL POWER above the limit of Required Action A.1. These Completion Times are intended to allow adequate time to increase THERMAL POWER to above the limit of Required Action A.1, while not permitting the core to remain with unconfirmed power distributions for extended periods of time.

Required Action A.6 is modified by a Note that states that the peaking factor surveillances may only be done after the excore detectors have been calibrated to show zero tilt (i.e., Required Action A.5). The intent of this Note is to have the peaking factor surveillances performed at operating power levels, which can only be accomplished after the excore detectors are calibrated to show zero tilt and the core returned to power.

B.1

If Required Actions A.1 through A.6 are not completed within their associated Completion Times, the unit must be brought to a MODE or condition in which the requirements do not apply. To achieve the status, THERMAL POWER must be reduced to < 50% RTP within 4 hours. The allowed Completion Time of 4 hours is reasonable based on operating experience regarding the amount of time required to reach the reduced power level without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1

SR 3.2.4.1 is modified by two Notes. Note 1 allows QPTR to be calculated with three power range channels if THERMAL POWER is < 75% RTP and the input from one Power Range Neutron Flux channel is inoperable. Note 2 allows performance of SR 3.2.4.2 in lieu of SR 3.2.4.1.

This Surveillance verifies that the QPTR as indicated by the Protection and Safety Monitoring System (PMS) excore channels is within its limits. The Frequency of 7 days takes into account other information and alarms available to the operator in the control room.

For those causes of QPT that occur quickly (a dropped rod), there are other indications of abnormality that prompt a verification of core power tilt.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.2.4.2

This Surveillance is modified by a Note, which states that it is not required until 12 hours after the input from one or more Power Range Neutron Flux channels are inoperable and the THERMAL POWER is ≥ 75% RTP.

With a PMS power range channel inoperable, tilt monitoring for a portion of the reactor core becomes degraded. Large tilts would likely be detected with the remaining channels, but the capability for detection of small power tilts in some quadrants is decreased. Performing SR 3.2.4.2 at a Frequency of 12 hours provides an accurate alternative means for assuring that any tilt remains within its limits.

For purposes of monitoring the QPTR when one power range channel is inoperable, the incore detectors are used to confirm that the normalized symmetric power distribution is acceptable.

With the OPDMS and one PMS channel inoperable, the surveillance of the incore power distribution on a 12 hour basis is sufficient to maintain peaking factors within their normal limits, especially, considering the other LCOs and ACTIONS required when the OPDMS is out of service.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors."

2. NUREG-0800, Standard Review Plan (SRP), Section 4.2, Revision 3, "Fuel System Design," (Appendix B).

3. Title 10, Code of Federal Regulations, Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," GDC 26, "Reactivity Control System Redundancy and Capability."

Deleted: Regulatory Guide 1.77, Rev. 0, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors," May 1974

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.5 OPDMS-Monitored Parameters

BASES

BACKGROUND

The On-line Power Distribution Monitoring System (OPDMS) for the AP1000 is an advanced core monitoring and support package. The OPDMS has the ability to continuously monitor core power distribution parameters. In addition, the OPDMS monitors SDM.

The purpose of the limits on the OPDMS-monitored power distribution parameters is to provide assurance of fuel integrity during Conditions I (Normal Operation) and II (incidents of Moderate Frequency) events by: (1) not exceeding the minimum departure from boiling ratio (DNBR) in the core, and (2) limiting the fission gas release, fuel pellet temperature, and cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the peak cladding temperature (PCT) limit of 2200°F is not exceeded.

The definition of certain quantities used in these specifications are as follows:

Peak kw/ft(Z) — Peak linear power density (axially dependent) as measured in kw/ft.

$F^N_{\Delta H}$ — Ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

Minimum DNBR — Minimum ratio of the critical heat flux to actual heat flux at any point in the reactor that is allowed in order to assure that certain performance and safety criteria requirements are met over the range of plant conditions.

By continuously monitoring the core and following its actual operation, it is possible to significantly limit the adverse nature of power distribution initial conditions for transients which may occur at any time.

APPLICABLE SAFETY ANALYSES

The limits on the above parameters preclude core power distributions from occurring which would violate the following fuel design criteria:

a. During a large break loss of coolant accident (LOCA), the PCT must not exceed a limit of 2200°F (Ref. 1);

b. During a loss of forced reactor coolant flow accident, there must be at least a 95% probability at a 95% confidence level (the 95/95 departure from nucleate boiling (DNB) criterion) that the hot fuel rod in the core does not experience a DNB condition;

c. During an ejected rod accident, the energy deposition to the fuel must not exceed applicable limits (Ref. 2); and

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d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn.

Limits on linear power density or peak kw/ft assure that the peak linear power density assumed as a base condition in the LOCA analyses is not exceeded during normal operation.

Limits on $F_{\Delta H}$ ensure that the LOCA analysis assumptions and assumptions made with respect to the Overtemperature ΔT Setpoint are maintained.

The limit on DNBR ensures that if transients analyzed in the safety analyses initiate from the conditions within the limit allowed by the OPDMS, the DNB criteria will be met.

The OPDMS-monitored power distribution parameters of this LCO satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO ensures operation within the bounds assumed in the safety analyses. Calculations are performed in the core design process to confirm that the core can be controlled in such a manner during operation that it can stay within these limits. If the LCO limits cannot be maintained within limits, reduction of the core power is required.

Violating the OPDMS-monitored power distribution parameter limits could result in unanalyzed conditions should a design basis event occur while the parameters are outside their specified limits.

LCO (continued)

Peak kw/ft limits define limiting values for core power peaking that precludes peak cladding temperatures above 2200°F during either a large or small break LOCA. The highest calculated linear power densities in the core at specific core elevations are displayed for operator visual verification relative to the COLR values.

The determination of $F_{\Delta H}^{N}$ identifies the coolant flow channel with the maximum enthalpy rise. This channel has the least heat removal capability and thus the highest probability for DNB. Should $F_{\Delta H}^{N}$ exceed the limit given in the COLR, the possibility exists for DNBR to exceed the value used as a base condition for the safety analysis.

Two levels of alarms on power distribution parameters are provided to the operator. One serves as a warning before the three parameters (kw/ft(Z), $F_{\Delta H}^{N}$, DNBR) exceed their values used as a base condition for the safety analysis. The other alarm indicates when the parameters have reached their limits.

APPLICABILITY

The OPDMS-monitored power distribution parameter limits must be maintained in MODE 1 above 50% RTP to preclude core power distributions from exceeding the limits assumed in the safety analyses. Applicability in other MODES, and MODE 1 below 50% RTP, is not required because there is either insufficient stored energy in the fuel or insufficient energy transferred to the reactor coolant to require a limit on the distribution of core power. The OPDMS monitoring of SDM is applicable in MODES 1 and 2 with $k_{eff} \geq 1.0$.

Specifically for $F_{\Delta H}^{N}$, the design bases accidents (DBAs) that are sensitive to $F_{\Delta H}^{N}$ in other MODES (MODES 2 through 5) have significant margin to DNB, and therefore, there is no need to restrict $F_{\Delta H}^{N}$ in these modes.

In addition to the alarms discussed in the LCO section above (alarms on OPDMS-monitored power distribution parameters), there is an alarm indicating the potential inoperability of the OPDMS itself.

Should the OPDMS be determined to be inoperable for other than reasons of alarms inoperable, this LCO is no longer applicable and LCOs 3.2.1 through 3.2.4 become applicable.

ACTIONS

A.1

With any of the OPDMS-monitored power distribution parameters outside of their limits, the assumptions used as most limiting base conditions for the DBA analyses may no longer be valid. The 1 hour operator ACTION requirement to restore the parameter to within limits is consistent with the basis for the anticipated operational occurrences and provides time to assess if there are instrumentation problems. It also allows the possibility to restore the parameter to within limits by rod cluster control assembly (RCCA) motion if this is possible. The OPDMS will continuously monitor these parameters and provide an indication when they are approaching their limits.

B.1

If the OPDMS-monitored power distribution parameters cannot be restored to within their limits within the Completion Time of ACTION A.1, it is likely that the problem is not due to a failure of instrumentation. Most of these parameters can be brought within their respective limits by reducing THERMAL POWER because this will reduce the absolute power density at any location in the core thus providing margin to the limit.

If the parameters cannot be returned to within limits as power is being reduced, THERMAL POWER must be reduced to < 50% RTP where the LCOs are no longer applicable.

A Note has been added to indicate that if the power distribution parameters in violation are returned to within their limits during the power reduction, then power operation may continue at the power level where this occurs. This is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions.

The Completion Time of 4 hours provides an acceptable time to reduce power in an orderly manner and without allowing the plant to remain outside the $F^N_{\Delta H}$ limits for an extended period of time.

C.1

If the SDM requirements are not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. It is assumed that boration will be continued until the SDM requirements are met. In the determination of the required combination of boration flow rate and boron concentration, there is no unique requirement that must be satisfied. Since it is imperative to raise the boron concentration of the RCS as soon

BASES

ACTIONS (continued)

as possible, the boron concentration should be a concentrated solution. The operator should begin boration with the best source available for the plant conditions.

SURVEILLANCE REQUIREMENTS

With OPDMS operating, the power distribution parameters are continuously computed and displayed, and compared against their limit. Two levels of alarms are provided to the operator. The first alarm provides a warning before these parameters (kw/ft(Z), $F^N_{\Delta H}$, and DNBR) exceed their limits. The second alarm indicates when they actually reach their limits. A third alarm indicates trouble with the OPDMS system.

SR 3.2.5.1

This Surveillance requires the operator to verify that the power distribution parameters are within their limits. This confirmation is a verification in addition to the automated checking performed by the OPDMS system. A 24 hour Surveillance interval provides assurance that the system is functioning properly and that the core limits are met.

With the OPDMS parameter alarms inoperable, an increased Surveillance Frequency is provided to assure that parameters are not approaching the limits. A 12 hour Frequency is adequate to identify changes in these parameters that could lead to their exceeding their limits.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors," 1974.

2. NUREG-0800, Standard Review Plan (SRP), Section 4.2, Revision 3, "Fuel System Design," (Appendix B).

Deleted: Regulatory Guide 1.77, Rev. 0, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors," May 1974.

APPENDIX D

CONFORMING CHANGES TO RELATED DCD CHAPTERS

Introduction

In addition to the Advanced First Core (AFC) changes to DCD Chapter 4 shown in Appendix A of this report and the revised accident analysis provided in DCD Chapter 15 as shown in Appendix B of this report, this section provides the conforming changes to DCD Chapters 1, 6 and 12 which are needed for overall consistency of the DCD.

Conforming Changes to DCD Tier 2 Chapter 1

Table 1.3-1, "PLANT COMPARISON WITH SIMILAR FACILITIES"

Table 1.3-1 is revised consistent with the enhanced GRCA design change with tungsten as the gray rod absorber material.

Table 1.3-1 (Sheet 1 of 6)

AP1000 PLANT COMPARISON WITH SIMILAR FACILITIES

Systems – Components	DCD	AP1000	AP600	Reference 2 Loop
Plant design objective	1.2	60 yrs	60 yrs	40 yrs
NSSS power	4.0	3,415 MWt	1,940 MWt	3,410 MWt
Core power	4.0	3,400 MWt	1,933 MWt	3,390 MWt
Net electrical output	1.2	≥ 1,000 MWe	600 MWe	1,075 MWe
Reactor operating pressure	5.1	2,250 psia	2,250 psia	2,250 psia
Hot leg temp	5.1	610°F	600°F	611°F (Cycle 1) 603°F (current)
Steam generator design pressure	5.4	1200 psia	1200 psia	1100 psia
Main feedwater temp	10.3	440°F	435°F	445°F
Core	4.0			
Number fuel assem.		157	145	217
Active fuel length		168 in	144 in	150 in
Fuel assembly array		17 x 17	17 x 17	16 x 16
Fuel rod OD		0.374 in	0.374 in	0.382 in
Number control assem.		53	45	83
– Absorber material		Ag-In-Cd	Ag-In-Cd	B_4C/Ag-In-Cd
Number gray rod assem.		16	16	8 (part length)
– Absorber material		Tungsten	SS-304/Ag-In-Cd	Inconel 625/ B_4C
Avg linear power		5.707 kW/ft	4.10 kW/ft	5.34 kW/ft
Heat flux hot channel factor, FQ		2.60	2.60	2.35

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Table 1.6-1, "MATERIAL REFERENCED"

The following Table 1.6-1 updates are provided consistent with the Westinghouse topical references changes in the DCD Chapters 4 and 15 mark-ups shown in Appendices A and B.

Table 1.6-1 (Sheet 5 of 21) MATERIAL REFERENCED		
DCD Section Number	**Westinghouse Topical Report Number**	**Title**
3H	[APP-GW-GLR-602	AP1000 Shield Building Design Details for Select Wall and RC/SC Connections, Revision 1, Westinghouse Electric Company LLC]*
4.1	WCAP-10444-P-A (P) WCAP-10445-NP-A	Reference Core Report VANTAGE 5 Fuel Assembly, September 1985, and VANTAGE 5H Fuel Assembly, Addendum 2A, February 1989
	WCAP-12610-P-A (P) WCAP-14342-A	VANTAGE+ Fuel Assembly Reference Core Report, April 1995
	[WCAP-12488-A (P) [WCAP-14204-A]*	Fuel Criteria Evaluation Process, October 1994]*
	WCAP-16943-P-A (P) WCAP-16943-NP-A	Enhanced GRCA Rodlet Design, September 2012
4.2	[WCAP-12488-A (P) [WCAP-14204-A]*	Fuel Criteria Evaluation Process, October 1994]*
	WCAP-10125-P-A (P) WCAP-10126-NP-A	Extended Burnup Evaluation of Westinghouse Fuel, December 1985
	WCAP-8183	Operational Experience with Westinghouse Cores
	WCAP-9179 (P) WCAP-9224	Properties of Fuel and Core Component Materials, July 1978
	WCAP-12610-P-A (P) WCAP-14342-A	VANTAGE+ Fuel Assembly Reference Core Report, June 1990/April 1995
	WCAP-8218-P-A (P) WCAP-8219-A	Fuel Densification Experimental Results and Model for Reactor Application, March 1975
	WCAP-10851-P-A (P) WCAP-11873-A	Improved Fuel Performance Models for Westinghouse Fuel Rod Design and Safety Evaluations, August 1988
	WCAP-13589-A (P) WCAP-14297-A	Assessment of Clad Flattening and Densification Power Spike Factor Elimination in Westinghouse Nuclear Fuel, March 1995
	WCAP-8963-P-A (P) WCAP-8964-A	Safety Analysis for the Revised Fuel Rod Internal Pressure Design Basis, August 1977
	WCAP-10021-P-A (P) WCAP-10377-NP-A	Westinghouse Wet Annular Burnable Absorber Evaluation Report, Revision 1, October 1983
	WCAP-10444-P-A (P) WCAP-10445-NP-A	Reference Core Report VANTAGE 5 Fuel Assembly, September 1985

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*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

Table 1.6-1 (Sheet 6 of 21) MATERIAL REFERENCED		
DCD Section Number	**Westinghouse Topical Report Number**	**Title**
4.2	WCAP-8278 (P) WCAP-8279	Hydraulic Flow Test of the 17x17 Fuel Assembly, February 1974
	WCAP-8691 (P) WCAP-8692	Fuel Rod Bow Evaluation, Revision 1, July 1979
	WCAP-9500-P-A (P) WCAP-9500-A	Reference Core Report 17x17 Optimized Fuel Assembly, May 1982
	WCAP-8236 (P) WCAP-8288	Safety Analysis of the 17x17 Fuel Assembly for Combined Seismic and Loss-of-Coolant Accident, December 1973
	WCAP-9401-P-A (P) WCAP-9402-A	Verification, Testing, and Analysis of the 17x17 Optimized Fuel Assembly, August 1981
	WCAP-9283	Integrity of Primary Piping Systems of Westinghouse Nuclear Power Plants During Postulated Seismic Events, March 1978
	WCAP-15063-P-A (P) WCAP-15064-NP-A	Westinghouse Improved Performance Analysis and Design Model (PAD 4.0), Revision 1, July 2000
	WCAP-8377 (P)	Revised Clad Flattening Model, July 1974
	WCAP-16652-P	AP1000 Core & Fuel Design Technical Report, Revision 0
	WCAP-16943-P-A (P) WCAP-16943-NP-A	Enhanced GRCA Rodlet Design, September 2012
	WCAP-8963-P-A, Addendum 1-A, Revision 1-A (P)	Safety Analysis for the Revised Fuel Rod Internal Pressure Design Basis (Departure from Nucleate Boiling Mechanistic Propagation Methodology), June 2006
	CENPD-404-P-A (P)	Implementation of ZIRLO™ Cladding Material in CE Nuclear Power Fuel Assembly Designs, November 2001
4.3	WCAP-9272-P-A (P) WCAP-9273-NP-A	Westinghouse Reload Safety Evaluation Methodology, July 1985
	[WCAP-12488-P-A (P) [WCAP-14204-A]*	Fuel Criteria Evaluation Process, October 1994]*
	WCAP-12472-P-A (P) WCAP-12473-A	BEACON: Core Monitoring and Operations Support System, August 1994; Addendum 1, May 1996; Addendum 2, March 2001
	WCAP-8330	Westinghouse Anticipated Transients Without Reactor Trip Analysis, August 1974
	WCAP-7308-L-P-A (P) WCAP-7308-L-A	Evaluation of Nuclear Hot Channel Factor Uncertainties, June 1988
	WCAP-8218-P-A (P) WCAP-8219-A	Fuel Densification Experimental Results and Model for Reactor Application, March 1975

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WCAP-7113 ... [1]

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Table 1.6-1 (Sheet 7 of 21) MATERIAL REFERENCED		
DCD Section Number	**Westinghouse Topical Report Number**	**Title**
4.3	WCAP-8359	Effects of Fuel Densification Power Spikes on Clad Thermal Transients, July 1974
	WCAP-7811	Power Distribution Control of Westinghouse Pressurized Water Reactors, December 1971
	WCAP-8385 (P) WCAP-8403	Power Distribution Control and Load Following Procedures, September 1974
	WCAP-10216-P-A (P) WCAP-10217-A	Relaxation of Constant Axial Offset Control, FQ Surveillance Technical Specification, Revision 1A, February 1994
	WCAP-7912-P-A (P) WCAP-7912-A	Power Peaking Factors, January 1975
	WCAP-9217 (P) WCAP-9218	Results of Control Rod Worth Program, October 1977
	WCAP-3696-8 (P)	Pressurized Water Reactor pH – Reactivity Effect Final Report, October 1968
	WCAP-3680-20 (P)	Xenon-Induced Spatial Instabilities in Large Pressurized Water Reactors, March 1968
	WCAP-3680-21 (P)	Control Procedures for Xenon-Induced X-Y Instabilities in Large Pressurized Water Reactors, February 1969
	WCAP-3680-22 (P)	Xenon-Induced Spatial Instabilities in Three Dimensions, September 1969
	WCAP-7964	Axial Xenon Transient Tests at the Rochester Gas and Electric Reactor, June 1971
	WCAP-7048-P-A (P) WCAP-7757-A	The PANDA Code, February 1975
	WCAP-7213-A (P) WCAP-7758-A	The TURTLE 24.0 Diffusion Depletion Code, February 1975
	WCAP-8768	Safety-Related Research and Development for Westinghouse Pressurized Water Reactors, Program Summaries – Winter 1977 – Summer 1978, Revision 2, October 1978

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WCAP-8498 ... [2]

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Table 1.6-1 (Sheet 8 of 21) MATERIAL REFERENCED		
DCD Section Number	**Westinghouse Topical Report Number**	**Title**
4.3	WCAP-6073 (P)	LASER – A Depletion Program for Lattice Calculations Based on MUFT and THERMOS, April 1966
	WCAP-2048 (P)	The Doppler Effect for a Non-Uniform Temperature Distribution in Reactor Fuel Elements, July 1962
	WCAP-11596-P-A (P) WCAP-11597-A	Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores, June 1988
	WCAP-10841 (P) WCAP-10842	Qualification of the PHOENIX/POLCA Nuclear Design and Analysis Program for Boiling Water Reactors, June 1985
	WCAP-7806	Nuclear Design of Westinghouse Pressurized Water Reactors with Burnable Poison Rods, December 1971
	WCAP-3385-56 Part II	Saxton Core II Fuel Performance Evaluation Part II: Evaluation of Mass Spectrometric and Radiochemical Analysis of Irradiated Saxton Plutonium Fuel, July 1973
	WCAP-3385-56 Part I	Saxton Core II - Fuel Performance Evaluation Part I: Materials, September 1971
	WCAP-3385-36	Saxton Plutonium Project - Quarterly Progress Report for the Period Ending June 20, 1973, July 1973
	WCAP-3385-37	Saxton Plutonium Project - Quarterly Progress Report for the Period Ending September 30, 1973, December 1973
	WCAP-3017-6094	Yankee Core Evaluation Program Final Report, January 1971
	WCAP-10965-P-A (P) WCAP-10966-A	ANC: A Westinghouse Advanced Nodal Computer Code, September 1986
	WCAP-3726-1	PuO2-UO2 Fueled Critical Experiments, July 1967
	WCAP-13589-A (P) WCAP-14297-A	Assessment of Clad Flattening and Densification Power Spike Factor Elimination in Westinghouse Nuclear Fuel, March 1995
	WCAP-13524 (P) WCAP-14952-NP-A	APOLLO - A One Dimensional Neutron Theory Program, Revision 1, August 1994
	WCAP-16652-P	AP1000 Core & Fuel Design Technical Report, Revision 0

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Table 1.6-1 (Sheet 9 of 21) MATERIAL REFERENCED		
DCD Section Number	**Westinghouse Topical Report Number**	**Title**
4.3	WCAP-3385-54	Saxton Plutonium Program Critical Experiments for the Saxton Partial Plutonium Core, Westinghouse Electric Corp., Atomic Power Division, December 1965
	WCAP-16045-P-A (P) WCAP-16045-NP-A	Qualification of the Two-Dimensional Transport Code PARAGON, August 2004
	WCAP-16943-P-A (P) WCAP-16943-NP-A	Enhanced GRCA Rodlet Design, September 2012
	WCAP-16045-P-A Addendum 1-A (P) WCAP-16045-NP-A Addendum 1-A	Qualification of the NEXUS Nuclear Data Methodology, August 2007
	WCAP-10965-P-A Addendum 2-A (P) WCAP-10966-A Addendum 2-A	Qualification of the New Pin Power Recovery Methodology, September 2010
4.4	WCAP-11397-P-A (P) WCAP-11397-A	Revised Thermal Design Procedure, April 1989
	WCAP-6065 (P)	Melting Point of Irradiated UO2, February 1965
	WCAP-10444-P-A (P) WCAP-10445-NP-A	Reference Core Report VANTAGE 5 Fuel Assembly, September 1985
	WCAP-9226-P (P) WCAP-9227-NP	Reactor Core Response to Excessive Secondary Steam Releases, January 1989
	WCAP-7695-L (P)	DNB Test Results for R-Grid Thimble Cold Wall Cells, Addendum 1, October 1972
	[WCAP-12488-A (P)	Westinghouse Fuel Criteria Evaluation Process, October 1994]*
	WCAP-7941-P-A (P) WCAP-7959-A	Effect of Axial Spacing on Interchannel Thermal Mixing with the R Mixing Vane Grid, January 1975
	WCAP-8298-P-A (P) WCAP-8290-A	The Effect of 17x17 Fuel Assembly Geometry on Interchannel Thermal Mixing, January 1975
	WCAP-8174 (P) WCAP-8202-A	Effect of Local Heat Flux Spikes on DNB in Non Uniform Heated Rod Bundles, August 1973
	WCAP-7667-P-A (P) WCAP-7755-A	Interchannel Thermal Mixing with Mixing Vane Grids, January 1975
	WCAP-8691 (P) WCAP-8692	Fuel Rod Bow Evaluation, Revision 1, July 1979
	WCAP-8054-P-A (P) WCAP-8195-A	Applications of THINC-IV Program to PWR Design, October 1973
	WCAP-7956-P-A (P)	THINC-IV, An Improved Program for Thermal-Hydraulic Analysis of Rod Bundle Cores, February 1989
	WCAP-2923	In-Pile Measurement of UO2 Thermal Conductivity, March 1966

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*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

Table 1.6-1 (Sheet 10 of 21) MATERIAL REFERENCED		
DCD Section Number	**Westinghouse Topical Report Number**	**Title**
4.4	WCAP-10851-P-A (P) WCAP-11873-A	Improved Fuel Performance Models for Westinghouse Fuel Rod Design and Safety Evaluations, August 1988
	WCAP-8720 Addendum 2	Revised PAD Code Thermal Safety Model, October 1982
	WCAP-6069	Burnup Physics of Heterogeneous Reactor Lattices, June 1965
	WCAP-3385-56 Part II	Saxton Core II Fuel Performance Evaluation: Evaluation of Mass Spectrometric and Radiochemical Analyses of Irradiated Saxton Plutonium Fuel, July 1970
	WCAP-7912-P-A (P) WCAP-7912-A	Power Peaking Factors, January 1975
	WCAP-8453-A	Analysis of Data from the Zion (Unit 1) THINC Verification Test, May 1976
	WCAP-12610-P-A (P) WCAP-14342-A	VANTAGE+ Fuel Assembly Reference Core Report, April 1995
	WCAP-15025-P-A (P) WCAP-15026-NP-A	Modified WRB-2 Correlation, WRB-2M, for Predicting Critical Heat Flux in 17x17 Rod Bundles with Modified LPD Mixing Vane Grids, April 1999
	WCAP-14565-P-A (P) WCAP-15306-NP-A	VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis, October 1999
	WCAP-15063-P-A (P) WCAP-15064-NP-A	Westinghouse Improved Performance Analysis and Design Model (PAD 4.0), Revision 1, July 2000
	WCAP-16652-NP	AP1000 Core & Fuel Design Technical Report, Revision 0
	WCAP-14565-P-A, Addendum 1-A (P)	Addendum 1 to WCAP-14565-P-A Qualification of ABB Critical Heat Flux Correlations with VIPRE-01 Code, August 2004
	WCAP-14565-P-A, Addendum 2-P-A (P)	Addendum 2 to WCAP-14565-P-A Extended Application of ABB-NV Correlation and Modified ABB-NV Correlation WLOP for PWR Low Pressure Applications, Revision 0, April 2008
5.2	WCAP-7907-P-A (P) WCAP-7907-A	LOFTRAN Code Description, April 1984
	WCAP-9292	Dynamic Fracture Toughness of ASME SA-508 Class 2a and ASME SA-533 Grade A Class 2 Base and Heat-Affected Zone Material and Applicable Weld Metals, March 1978
	WCAP-7477-L (P) WCAP-7735	Sensitized Stainless Steel in Westinghouse PWR Nuclear Steam Supply Systems, March 1970 (P), August 1971 (Non-P)

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Table 1.6-1 (Sheet 14 of 21) MATERIAL REFERENCED		
DCD Section Number	**Westinghouse Topical Report Number**	**Title**
13	WCAP-13864	Rod Control System Evaluation Program, Revision 1-A, November 1994
15.0	WCAP-11397-P-A (P) WCAP-11397-A	Revised Thermal Design Procedure, April 1989
	WCAP-10054-P-A (P) WCAP-10081	Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code, August 1985
	WCAP-12945-P-A (P) WCAP-14747	Code Qualification Document for Best Estimate LOCA Analysis, Revision 1, March 1998
	WCAP-7908-A	FACTRAN – A FORTRAN-IV Code for Thermal Transients in a UO2 Fuel Rod, December 1989
	WCAP-7907-P-A (P) WCAP-7907-A	LOFTRAN Code Description, April 1984
	WCAP-7979-P-A (P) WCAP-8028-A	TWINKLE – A Multi-Dimensional Neutron Kinetics Computer Code, January 1975
	WCAP-10698-P-A (P) WCAP-10750-A	SGTR Analysis Methodology to Determine the Margin to Steam Generator Overfill, August 1987
	WCAP-14234 (P) WCAP-14235	LOFTRAN and LOFTTR2 AP600 Code Applicability Document, Revision 1, August 1997
	WCAP-15644-P (P) WCAP-15644-NP	AP1000 Code Applicability Report, Revision 2, March 2004
	WCAP-16009-P-A (P) WCAP-16009-NP-A	Realistic Large Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)
15.1	WCAP-7907-P-A (P) WCAP-7907-A	LOFTRAN Code Description, April 1984
	WCAP-11397-P-A (P) WCAP-11397-A	Revised Thermal Design Procedure, April 1989
	WCAP-9226 (P) WCAP-9227	Reactor Core Response to Excessive Secondary Steam Releases, Revision 1, February 1998
	WCAP-7908-A	FACTRAN – A FORTRAN-IV Code for Thermal Transients in a UO2 Fuel Rod, December 1989
	WCAP-15644-NP	AP1000 Code Applicability Report, Revision 2, March 2004
	WCAP-14565-P-A (P) WCAP-15306-NP-A	VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal Hydraulic Safety Analysis, October 1999

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Table 1.6-1 (Sheet 15 of 21) MATERIAL REFERENCED		
DCD Section Number	**Westinghouse Topical Report Number**	**Title**
15.2	WCAP-7769	Overpressure Protection for Westinghouse Pressurized Water Reactors, Revision 1, June 1972
	WCAP-16779-NP	Overpressure Protection Report for AP1000 Nuclear Power Plant
	WCAP-7907-P-A (P) WCAP-7907-A	LOFTRAN Code Description, April 1984
	WCAP-9230 (P) WCAP-9231	Report on the Consequences of a Postulated Main Feedline Rupture, January 1978
	WCAP-11397-P-A (P) WCAP-11397-A	Revised Thermal Design Procedure, April 1989
	WCAP-15644-P (P) WCAP-15644-NP	AP1000 Code Applicability Report, Revision 2, March 2004
	WCAP-7908-A	FACTRAN - A FORTRAN-IV Code for Thermal Transients in a UO2 Fuel Rod, December 1989
	WCAP-14565-P-A (P) WCAP-15306-NP-A	VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal Hydraulic Safety Analysis, October 1999
15.3	WCAP-7907-P-A (P) WCAP-7907-A	LOFTRAN Code Description, April 1984
	WCAP-7908-A	FACTRAN - A FORTRAN-IV Code for Thermal Transients in a UO2 Fuel Rod, December 1989
	WCAP-8424	An Evaluation of Loss of Flow Accidents Caused by Power System Frequency Transients in Westinghouse PWRs, Revision 1, May 1975
	WCAP-11397-P-A (P) WCAP-11397-A	Revised Thermal Design Procedure, April 1989
	WCAP-14565-P-A (P) WCAP-15306-NP-A	VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal Hydraulic Safety Analysis, October 1999
	WCAP-15644-P (P) WCAP-15644-NP	AP1000 Code Applicability Report, Revision 2, March 2004
15.4	WCAP-7979-P-A (P) WCAP-8028-A	TWINKLE - A Multi-Dimensional Neutron Kinetics Computer Code, January 1975
	WCAP-7908-A	FACTRAN - A FORTRAN-IV Code for Thermal Transients in a UO2 Fuel Rod, December 1989
	WCAP-7907-P-A (P) WCAP-7907-A	LOFTRAN Code Description, April 1984

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Table 1.6-1 (Sheet 16 of 21) MATERIAL REFERENCED		
DCD Section Number	**Westinghouse Topical Report Number**	**Title**
15.4	WCAP-15806-P-A (P) WCAP-15807-NP-A	Westinghouse Control Rod Ejection Accident Analysis Methodology Using Multi-Dimensional Kinetics
	WCAP-10965-P-A (P) WCAP-10966-A	ANC: A Westinghouse Advanced Nodal Computer Code, September 1986
	WCAP-11397-P-A (P) WCAP-11397-A	Revised Thermal Design Procedure, April 1989
	WCAP-15644-P (P) WCAP-15644-NP	AP1000 Code Applicability Report, Revision 2, March 2004
	WCAP-11596-P-A (P) WCAP-11597-A	Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores, June 1988
	WCAP-16045-P-A (P) WCAP-16045-NP-A	Qualification of the Two-Dimensional Transport Code PARAGON, August 2004
	WCAP-10965-P-A, Addendum 1 (P) WCAP-10966-A Addendum 1	ANC – A Westinghouse Advanced Nodal Computer Code; Enhancements to ANC Rod Power Recovery, April 1989
	WCAP-14565-P-A (P) WCAP-15306-NP-A	VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis, October 1999
	WCAP-15063-P-A, Revision 1 with Errata (P) WCAP-15064-NP-A	Westinghouse Improved Performance Analysis and Design Model (PAD 4.0), July 2000
	WCAP-16045-P-A Addendum 1-A (P) WCAP-16045-NP-A Addendum 1-A	Qualification of the NEXUS Nuclear Data Methodology, August, 2007
	WCAP-10965-P-A, Addendum 2-A (P)	Qualification of the New Pin Power Recovery Methodology, September, 2010
	WCAP-15025-P-A (P) WCAP-15026-NP-A	Modified WRB-2 Correlation, WRB-2M, for Predicting Critical Heat Flux in 17x17 Rod Bundles with Modified LPD Mixing Vane Grids, April 1999
15.5	WCAP-7907-P-A (P) WCAP-7907-A	LOFTRAN Code Description, April 1984
	WCAP-14565-P-A (P) WCAP-15306-NP-A	VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis, October 1999

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Table 1.6-1 (Sheet 16a of 21) MATERIAL REFERENCED		
DCD Section Number	**Westinghouse Topical Report Number**	**Title**
15.6		Letter from R. C. Jones, Jr. (USNRC), to N. J. Liparulo (W), Subject: Acceptance for Referencing of the Topical Report, WCAP-12945 (P), Westinghouse CQD for Best Estimate LOCA Analysis, June 28, 1996
	WCAP-12945-P-A (P) WCAP-14747	Code Qualification Document for Best Estimate Analysis, Revision 2, March 1998
	WCAP-10079-P-A (P) WCAP-10080-A	NOTRUMP – A Nodal Transient Small Break and General Network Code, August 1985
	WCAP-10054-P-A (P) WCAP-10081-A	Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code, August 1985
	WCAP-7907-P-A (P) WCAP-7907-A	LOFTRAN Code Description, April 1984
	WCAP-7908-A	FACTRAN – A FORTRAN-IV Code for Thermal Transients in a UO_2 Fuel Rod, December 1989
	WCAP-11397-P-A (P) WCAP-11397-A	Revised Thermal Design Procedure, April 1989
	WCAP-10698-P-A (P) WCAP-10750-A	SGTR Analysis Methodology to Determine the Margin to Steam Generator Overfill, August 1987
	WCAP-14206 (P) WCAP-14207	Applicability of the NOTRUMP Computer Code to AP600 SSAR Small-Break LOCA Analyses, November 1994

Table 1.6-1 (Sheet 17 of 21) MATERIAL REFERENCED		
DCD Section Number	**Westinghouse Topical Report Number**	**Title**
15.6	WCAP-14601 (P) WCAP-15062	AP600 Accident Analyses – Evaluation Models, Revision 2, May 1998
	WCAP-14234 (P) WCAP-14235	LOFTRAN and LOFTTR2 AP600 Code Applicability Document, Revision 1, August 1997
	WCAP-14171 (P) WCAP-14172	WCOBRA/TRAC Applicability to AP600 Large-Break Loss-of-Coolant Accident, Revision 2, March 1998
	WCAP-14807 (P) WCAP-14808	NOTRUMP Final Validation Report for AP600, Revision 5, August 1998
	WCAP-14776 (P) WCAP-14777	WCOBRA/TRAC OSU Long-Term Cooling Final Validation Report, Revision 4, April 1998
	WCAP-15644-P (P) WCAP-15644-NP	AP1000 Code Applicability Report, Revision 2, March 2004
	WCAP-15613 (P) WCAP-15706	AP1000 PIRT and Scaling Assessment, March 2001
	WCAP-16009-P-A (P) WCAP-16009-NP-A	Realistic Large Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)
	WCAP-14449-P-A (P) WCAP-14450	Application of Best Estimate Large Break LOCA Methodology to Westinghouse PWRs with Upper Plenum Injection, Revision 1
16.1	WCAP-9272-P-A (P) WCAP-9273-NP-A	Westinghouse Reload Safety Evaluation Methodology, July 1985
	WCAP-11618	Methodically Engineered, Restructured and Improved, Technical Specifications, Merits Program – Phase II Task 5 Criteria Application, November 1987, including Addendum 1, April 1989
	WCAP-8385 (P) WCAP-8403	Power Distribution Control and Load Following Procedures, September 1974
	WCAP-10216-P-A (P) WCAP-10217-A	Relaxation of Constant Axial Offset Control F_Q Surveillance Technical Specifications, Revision 1A, February 1994
	WCAP-12945-P-A (P) WCAP-14747	Code Qualification Document for Best Estimate Loss of Coolant Accident Analysis, Revision 1, March 1998

(P) Denotes Document is Proprietary

Table 1.9-1, "REGULATORY GUIDE / DCD SECTION CROSS-REFERENCES"

Table 1.9-1, DCD Section Cross-Reference for Regulatory Guide 1.77, is revised for consistency with NUREG-1793 Supplement 2 (DCD Revision 19 FSER) 15.2.4.8.2, "Staff Position Related to the Revision of NUREG-0800 Section 15.4.8".

Table 1.9-1 (Sheet of 7 of 15)

REGULATORY GUIDE/DCD SECTION CROSS-REFERENCES

Division 1 Regulatory Guide		DCD Chapter, Section or Subsection
1.77	Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors (Rev. 0, May 1974)	Regulatory Guide 1.77 is superseded by: The interim acceptance criteria specified in Appendix B to NUREG-0800 Section 4.2, Revision 3, will be followed for the analysis of the RCCA ejection accident, in accordance with NUREG-1793 Supplement 2, Section 15.2.4.8.2. The guidance of Regulatory Guide 1.183, "Alternative Radiological Source Terms For Evaluating Design Basis Accidents At Nuclear Power Reactors" will be followed instead of Reg. Guide 1.77.

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APPENDIX 1A

CONFORMANCE WITH REGULATORY GUIDES

Criteria Section	Referenced Criteria	AP1000 Position	Clarification/Summary Description of Exceptions
Reg. Guide 1.77, Rev. 0, 5/74 – Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors			
General		N/A	The interim acceptance criteria specified in Appendix B to NUREG-0800 Section 4.2, Revision 3, will be followed for the analysis of the RCCA ejection accident instead of Reg. Guide 1.77. The guidance of Reg. Guide 1.183, "Alternative Radiological Source Terms For Evaluating Design Basis Accidents At Nuclear Power Reactors" will be followed instead of Reg. Guide 1.77.

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Conforming Changes to DCD Tier 2 Chapter 6

DCD Section 6.4.4 is revised for consistency with the revised DCD Chapter 15 analyses. The rod ejection radiological dose analysis (Section 15.4.8) for the Advanced First Core has been recalculated to support the revised fission product release requirements and guidance of SRP Section 4.2, Revision 3, Appendix B, as modified by draft Regulatory Guide DG-1199 and subsequent clarification provided in ML111890397.

DCD Section 6.4.4 "System Safety Evaluation"

In the event of an accident involving the release of radioactivity to the environment, the nuclear island nonradioactive ventilation system (VBS) is expected to switch from the normal operating mode to the supplemental air filtration mode to protect the main control room personnel. Although the VBS is not a safety-related system, it is expected to be available to provide the necessary protection for realistic events. However, the design basis accident doses reported in Chapter 15 utilize conservative assumptions, and the main control room doses are calculated based on operation of the safety-related emergency habitability system (VES) since this is the system that is relied upon to limit the amount of activity the personnel are exposed to. The analyses assume that the VBS is initially in operation, but fails to enter the supplemental air filtration mode on a High-1 radioactivity indication in the main control room atmosphere. VES operation is then assumed to be initiated once the High-2 level for control room atmosphere activity is reached.

Doses are also calculated assuming that the VBS does operate in the supplemental air filtration mode as designed, but with no switchover to VES operation. This VBS operating case demonstrates the defense-in-depth that is provided by the system and also shows that, in the event of an accident with realistic assumptions, the VBS is adequate to protect the control room operators without depending on VES operation.

Doses were determined for the following design basis:

	VES Operating	VBS Operating
Large Break LOCA	< 5 rem TEDE (TBD)	< 5 rem TEDE (TBD)
Fuel Handling Accident	< 5 rem TEDE (TBD)	< 5 rem TEDE (TBD)
Steam Generator Tube Rupture		
(Pre-existing iodine spike)	< 5 rem TEDE (TBD)	< 5 rem TEDE (TBD)
(Accident-initiated iodine spike)	< 5 rem TEDE (TBD)	< 5 rem TEDE (TBD)
Steam Line Break		
(Pre-existing iodine spike)	< 5 rem TEDE (TBD)	< 5 rem TEDE (TBD)
(Accident-initiated iodine spike)	< 5 rem TEDE (TBD)	< 5 rem TEDE (TBD)
Rod Ejection Accident	< 5 rem TEDE (TBD)	< 5 rem TEDE (TBD)
Locked Rotor Accident		
(Accident without feedwater available)	< 5 rem TEDE (TBD)	< 5 rem TEDE (TBD)
(Accident with feedwater available)	< 5 rem TEDE (TBD)	< 5 rem TEDE (TBD)
Small Line Break Outside Containment	< 5 rem TEDE (TBD)	< 5 rem TEDE (TBD)

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Conforming Changes to DCD Tier 2 Chapter 12

The following changes to DCD Chapter 12 Section 12.2.1.2.4 and Tables 12.2-15 through 18 are provided for consistency with the enhanced GRCA design described in Chapter 4. Radiation sources associated with the gray rods were re-analyzed based on the associated material and dimensional changes.

Section 12.2.1.2.4 "Irradiated Control Rods, Gray Rods, and Secondary Source Rods"

The gamma ray source strengths of the irradiated control rods, gray rods, and secondary source rods are used in establishing radiation shielding requirements during refueling operations and during shipping of irradiated rods.

The absorber material used in the control rods is silver-indium-cadmium (Ag-In-Cd). The gray rods contain tungsten in a nickel alloy 718 sleeve. The gamma ray source strengths associated with the irradiated Ag-In-Cd control rod absorber and gray rod materials are listed in Table 12.2-15 for various times after shutdown.

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The photoneutron source material used in the secondary source rods is an equal volume mixture of antimony and beryllium (Sb-Be). The gamma ray source strengths associated with the secondary source rods are listed in Table 12.2-16 for various times after shutdown and Table 12.2-17 lists the neutron source strengths. The source values are per cubic centimeter of source material for an irradiation period of 400 days.

The material used for the control rod cladding, gray rod cladding and/or pellets and secondary source rod cladding is Type 304 stainless steel with a maximum cobalt content of 0.05 weight percent. The gamma ray source strengths associated with the irradiated stainless steel are listed in Table 12.2-18 for various times after shutdown.

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Table 12.2-15

IRRADIATED
CONTROL ROD AND GRAY ROD SOURCE STRENGTHS

Energy Group (Mev/gamma)	Silver-Indium-Cadmium Control Rod Source Strength at Time After Shutdown (Mev/cm^3-sec)		
	1 Day	**1 Week**	**1 Month**
0.20 - 0.40	2.3E+08	2.3E+08	2.2E+08
0.40 - 0.90	1.1E+12	1.1E+12	1.0E+12
0.90 - 1.35	2.0E+11	1.9E+11	1.8E+11
1.35 - 1.80	3.7E+11	3.7E+11	3.4E+11
(Mev/gamma)	**6 Months**	**1 Year**	**5 Years**
0.20 - 0.40	1.4E+08	8.5E+07	1.5E+06
0.40 - 0.90	6.6E+11	4.0E+11	7.1E+09
0.90 - 1.35	1.2E+11	7.2E+10	1.3E+09
1.35 - 1.80	2.3E+11	1.4E+11	2.5E+09

Note: The absorber cross-sectional area is 0.589 and 0.521 square centimeters for the upper and lower portions of the rods, respectively, and the absorber material density is 10.2 grams per cubic centimeter.

Energy Group (Mev/gamma)	Tungsten Gray Rod Source Strength at Time After Shutdown (Mev/cm^3-sec)		
	1 Day	**1 Week**	**1 Month**
0.00 - 0.25	6.5E+11	2.7E+11	8.3E+10
0.25 - 0.40	1.2E+11	5.2E+10	4.1E+10
0.40 - 0.90	5.1E+11	1.3E+11	1.1E+11
0.90 - 1.35	3.7E+10	7.9E+09	7.4E+09
1.35 - 1.80	9.2E+09	4.5E+07	8.7E+06
1.80 - 2.20	2.0E+09	7.2E+06	4.7E+05
(Mev/gamma)	**6 Months**	**1 Year**	**5 Years**
0.00 - 0.25	8.1E+09	1.7E+09	1.9E+06
0.25 - 0.40	2.1E+10	9.8E+09	2.6E+07
0.40 - 0.90	5.5E+10	2.5E+10	6.8E+07
0.90 - 1.35	5.5E+09	4.3E+09	2.1E+09
1.35 - 1.80	3.2E+06	1.0E+06	2.6E+03
1.80 - 2.20	1.3E+05	5.0E+04	2.1E+04

Note: The absorber cross-sectional area is 0.197 square centimeters and the absorber material density is 19.2 grams per cubic centimeter.

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Table 12.2-15 (continued) **IRRADIATED CONTROL ROD AND GRAY ROD SOURCE STRENGTHS**			
Energy Group (Mev/gamma)	**Gray Rod Sleeve (Nickel Alloy 718) Source Strength at Time After Shutdown (Mev/cm^3-sec)**		
	1 Day	**1 Week**	**1 Month**
0.00 - 0.25	8.2E+09	2.9E+09	1.3E+09
0.25 - 0.40	1.3E+10	1.1E+10	5.9E+09
0.40 - 0.90	4.1E+11	3.6E+11	2.4E+11
0.90 - 1.35	2.1E+12	2.1E+12	2.1E+12
1.35 - 1.80	8.0E+08	5.6E+08	4.4E+08
1.80 - 2.20	1.9E+08	2.0E+07	2.0E+07
2.20 - 2.60	7.9E+06	4.2E+04	4.1E+04
2.60 - 3.00	1.0E+08	1.0E+05	-
(Mev/gamma)	**6 Months**	**1 Year**	**5 Years**
0.00 - 0.25	9.4E+08	8.4E+08	4.9E+08
0.25 - 0.40	1.8E+08	3.1E+07	1.6E+07
0.40 - 0.90	2.5E+10	5.1E+09	4.5E+08
0.90 - 1.35	2.0E+12	1.8E+12	1.1E+12
1.35 - 1.80	1.0E+08	1.6E+07	4.4E+02
1.80 - 2.20	1.9E+07	1.8E+07	1.0E+07
2.20 - 2.60	3.9E+04	3.7E+04	2.2E+04
2.60 - 3.00	-	-	-

Note: The absorber cross-sectional area is 0.29 square centimeters and the material density is 8.2 grams per cubic centimeter.

Table 12.2-16

IRRADIATED SB-BE SECONDARY SOURCE ROD GAMMA RAY SOURCE STRENGTHS

Energy Group (Mev/gamma)	**Source Strength at Time After Shutdown (Mev/watt-sec)**		
	1 Day	**1 Week**	**1 Month**
0.20 - 0.40	3.0E+10	2.9E+10	2.5E+10
0.40 - 0.90	1.1E+13	7.0E+12	4.6E+12
0.90 - 1.35	6.7E+11	4.8E+11	3.4E+11
1.35 - 1.80	7.6E+12	7.1E+12	5.5E+12
1.80 - 2.20	9.8E+11	9.1E+11	7.0E+11
(Mev/gamma)	**6 Months**	**1 Year**	**5 Years**
0.20 - 0.40	1.1E+10	3.7E+09	2.2E+07
0.40 - 0.90	8.1E+11	9.7E+10	1.8E+08
0.90 - 1.35	6.0E+10	7.0E+09	
1.35 - 1.80	9.7E+11	1.2E+11	
1.80 - 2.20	1.2E+11	1.5E+10	

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Notes:

The Sb-Be material density is 3.56 grams per cubic centimeter.

The secondary source rod cross-sectional area is 0.432 square centimeter per rod.

The average neutron energy is 30 kev.

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Table 12.2-17 **IRRADIATED SB-BE SECONDARY SOURCE ROD NEUTRON SOURCE STRENGTHS**		
Time After Shutdown	**Sb-124 Concentration (curies/cm^3)**	**Neutron Source Strength (n/cm^3-sec)**
1 day	230	4.5E+08
1 week	210	4.2E+08
1 month	160	3.2E+08
6 months	29	5.8E+07
1 year	3.4	6.8E+06
5 years	0	0

Notes:
The Sb-Be material density is 3.56 grams per cubic centimeter.
The secondary source rod cross-sectional area is 0.432 square centimeter per rod.
The average neutron energy is 30 kev.

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Table 12.2-18

IRRADIATED STAINLESS STEEL SOURCE STRENGTHS (0.05 WEIGHT PERCENT COBALT)

Energy Group (Mev/gamma)	Source Strength at Time After Shutdown (Mev/cm³-sec)		
	1 Day	**1 Week**	**1 Month**
0.00 - 0.25	4.7E+08	4.2E+08	3.2E+08
0.25 - 0.40	1.1E+10	9.6E+09	5.4E+09
0.40 - 0.90	3.7E+10	3.5E+10	3.2E+10
0.90 - 1.35	2.3E+11	2.2E+11	2.1E+11
1.35 - 1.80	3.6E+08	1.3E+08	1.0E+08
1.80 - 2.20			
2.20 - 2.60			
(Mev/gamma)	**6 Months**	**1 Year**	**5 Years**
0.00 - 0.25	1.1E+08	7.8E+07	4.4E+07
0.25 - 0.40	1.3E+08	4.2E+06	1.5E+06
0.40 - 0.90	1.8E+10	1.1E+10	4.3E+08
0.90 - 1.35	1.8E+11	1.6E+11	9.7E+10
1.35 - 1.80	2.1E+07	3.2E+06	4.0E+01
1.80 - 2.20	1.7E+06	1.6E+06	9.3E+05
2.20 - 2.60	3.5E+10	3.3E+03	1.9E+03

Notes:
The various cross-section areas per rod are as follows:
- Ag-In-Cd control rod cladding - 0.136 cm^2
- Sb-Be secondary source rod cladding - 0.136 cm^2 (outer) and 0.118 cm^2 (inner)
- Gray rod cladding - 0.184 cm^2

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REFERENCES

D-1 DRAFT REGULATORY GUIDE DG-1199 (Proposed Revision 1 of Regulatory Guide 1.183, dated July 2000) "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors" (ML090960464, Publish Date 10/2009)

D-2 ML111890397 - TECHNICAL BASIS FOR REVISED REGULATORY GUIDE 1.183 (DG-1199) FISSION PRODUCT FUEL-TO-CLADDING GAP INVENTORY

APPENDIX E

CHAPTER 4

REACTOR

4.1 Summary Description

This chapter describes the mechanical components of the reactor and reactor core, including the fuel rods and fuel assemblies, the nuclear design, and the thermal-hydraulic design.

The reactor contains a matrix of fuel rods assembled into mechanically identical fuel assemblies along with control and structural elements. The assemblies, containing various fuel enrichments, are configured into the core arrangement located and supported by the reactor internals. The reactor internals also direct the flow of the coolant past the fuel rods. The coolant and moderator are light water at a normal operating pressure of 2250 psia. The fuel, internals, and coolant are contained within a heavy walled reactor pressure vessel. An AP1000 fuel assembly consists of 264 fuel rods in a 17x17 square array. The center position in the fuel assembly has a guide thimble that is reserved for in-core instrumentation. The remaining 24 positions in the fuel assembly have guide thimbles. The guide thimbles are joined to the top and bottom nozzles of the fuel assembly and provide the supporting structure for the fuel grids.

The fuel grids consist of an egg-crate arrangement of interlocked straps that maintain lateral spacing between the rods. The grid straps have spring fingers and dimples that grip and support the fuel rods. The intermediate mixing vane grids also have coolant mixing vanes. In addition, there are four intermediate flow mixing (IFM) grids. The IFM grid straps contain support dimples and coolant mixing vanes only. The top and bottom grids and protective grid do not contain mixing vanes.

The AP1000 fuel assemblies are similar to the 17x17 Robust and 17x17 XL Robust fuel assemblies. The 17x17 Robust fuel assemblies have an active fuel length of 12 feet and three intermediate flow mixing grids in the top mixing vane grid spans. The 17x17 XL Robust fuel assemblies have an active fuel length of 14 feet (same as AP1000 fuel assemblies) and no intermediate flow mixing grids. The AP1000 fuel assemblies have four intermediate flow mixing grids in the top mixing vane grid spans.

There is substantial operating experience with the 17x17 Robust and 17x17 XL Robust fuel assemblies. The 17x17 Robust fuel assemblies are described in References 1, 2 and 3. The 17x17 XL Robust fuel assemblies are described in References 4 and 5.

The XL Robust fuel assembly evolved from the previous VANTAGE+, VANTAGE 5 and VANTAGE 5 HYBRID designs. The XL Robust fuel assembly is based on the substantial

design and operating experience with those designs. The design is described and evaluated in References 2, 3, 6 through 10.

A number of proven design features have been incorporated in the AP1000 fuel assembly design. The AP1000 fuel assembly design includes: low pressure drop intermediate grids, four intermediate flow mixing (IFM) grids, a reconstitutable Westinghouse integral nozzle (WIN), and extended burnup capability. The bottom nozzle is a debris filter bottom nozzle (DFBN) that minimizes the potential for fuel damage due to debris in the reactor coolant. The AP1000 fuel assembly design also includes a protective grid for enhanced debris resistance.

The fuel rods consist of enriched uranium, in the form of cylindrical pellets of uranium dioxide, contained in ZIRLO®[1] (Reference 8) tubing. The tubing is plugged and seal welded at the ends to encapsulate the fuel. An axial blanket comprised of fuel pellets with reduced enrichment may be placed at each end of the enriched fuel pellet stack to reduce the neutron leakage and to improve fuel utilization.

Other types of fuel rods may be used to varying degrees within some fuel assemblies. One type uses an integral fuel burnable absorber (IFBA) containing a thin boride coating on the surface of the fuel pellets. The boride-coated fuel pellets provide a burnable absorber integral to the fuel.

Fuel rods are pressurized internally with helium during fabrication to reduce clad creepdown during operation and thereby prevent clad flattening. The fuel rods in the AP1000 fuel assemblies contain additional gas space below the fuel pellets, compared to the 17x17 Robust, 17x17 XL Robust and other previous fuel assembly designs to allow for increased fission gas production due to high fuel burnups.

Depending on the position of the assembly in the core, the guide thimbles are used for rod cluster control assemblies (RCCAs), gray rod cluster assemblies (GRCAs), neutron source assemblies, wet annular burnable absorber (WABA) assemblies, or thimble plugs.

For the initial core design, WABAs and IFBAs are used. Discrete burnable absorber designs, integral fuel burnable absorber designs or combinations may be used in subsequent reloads.

The bottom nozzle is a box-like structure that serves as the lower structural element of the fuel assembly and directs the coolant flow distribution to the assembly. The size of flow passages through the bottom nozzle limits the size of debris that can enter the fuel assembly.

1 ZIRLO® is a registered trademark in the United States of Westinghouse Electric Company LLC, its subsidiaries and/or its affiliates. This mark may also be used and/or registered in other countries throughout the world. All rights reserved. Unauthorized use is strictly prohibited. Other names may be trademarks of their respective owners.

The top nozzle assembly serves as the upper structural element of the fuel assembly and provides a partial protective housing for the rod cluster control assembly or other components.

The rod cluster control assemblies consist of 24 absorber rodlets fastened at the top end to a common hub, or spider assembly. Each absorber rod consists of an alloy of silver-indium-cadmium, which is clad in stainless steel. The rod cluster control assemblies are used to control relatively rapid changes in reactivity and to control the axial power distribution.

The gray rod cluster assemblies consist of 24 rodlets fastened at the top end to a common hub, or spider assembly (Reference 11). Geometrically, the gray rod cluster assembly is the same as a rod cluster control assembly except that the absorber rods consists of tungsten contained within a nickel-chromium-iron Alloy 718 sleeve. Stainless steel spacers are provided at the bottom of the rodlet between the bottom of the sleeve and the bottom end plug of the stainless steel cladding.

The gray rod cluster assemblies are used in base load operation and load follow maneuvering. The assemblies provide a mechanical shim reactivity mechanism to minimize the need for changes to the concentration of soluble boron.

The reactor core is cooled and moderated by light water at a pressure of 2250 psia. Soluble boron in the moderator/coolant serves as a neutron absorber. The concentration of boron is varied to control reactivity changes that occur relatively slowly, including the effects of fuel burnup. Burnable absorbers are also employed in the initial cycle to limit the amount of soluble boron required and, thereby maintain the desired negative reactivity coefficients.

The nuclear design analyses establish the core locations for control rods and burnable absorbers. The analyses define design parameters, such as fuel enrichments and boron concentration in the coolant.

The nuclear design establishes that the reactor core and the reactor control system satisfy design criteria, even if the rod cluster control assembly of the highest reactivity worth is in the fully withdrawn position.

The core has inherent stability against diametral and azimuthal power oscillations. Axial power oscillations, which may be induced by load changes, and resultant transient xenon may be suppressed by the use of the rod cluster control assemblies.

The control rod drive mechanisms used to withdraw and insert the rod cluster control assemblies and the gray rod cluster assemblies are described in subsection 3.9.4.

The thermal-hydraulic design analyses establish that adequate heat transfer is provided between the fuel clad and the reactor coolant. The thermal design takes into account local variations in dimensions, power generation, flow distribution, and mixing. The mixing vanes incorporated in the fuel assembly spacer grid design and the fuel assembly intermediate flow mixers induce additional flow mixing between the various flow channels within a fuel assembly, as well as between adjacent assemblies.

The reactor internals direct the flow of coolant to and from the fuel assemblies and are described in subsection 3.9.5.

The performance of the core is monitored by fixed neutron detectors outside the core, fixed neutron detectors within the core, and thermocouples at the outlet of selected fuel assemblies. The ex-core nuclear instrumentation provides input to automatic control functions.

Table 4.1-1 presents a summary of the principal nuclear, thermal-hydraulic, and mechanical design parameters of the AP1000 fuel. A comparison is provided to the fuel design used in AP1000, AP600 and in a licensed Westinghouse-designed plant using XL Robust fuel. For the comparison with a plant containing XL Robust fuel, a 193 fuel assembly plant is used, since no domestic, Westinghouse-designed 157 fuel assembly plants use 17x17 XL Robust fuel.

Table 4.1-2 tabulates the analytical techniques employed in the core design. The design basis must be met using these analytical techniques. Enhancements may be made to these techniques provided that the changes are bounded by NRC-approved methods, models, or criteria. In addition, application of the process described in WCAP-12488-A, (Reference 9) allows the Combined License holder to make fuel mechanical changes. Table 4.1-3 tabulates the mechanical loading conditions considered for the core internals and components. Specific or limiting loads considered for design purposes of the various components are listed as follows: fuel assemblies in subsection 4.2.1.5; control rods (RCCAs and GRCAs), burnable absorber rods, and neutron source rods, in subsection 4.2.1.6. The dynamic analyses, input forcing functions, and response loadings for the control rod drive system and reactor vessel internals are presented in subsections 3.9.4 and 3.9.5.

4.1.1 Principal Design Requirements

The fuel and control rod mechanism are designed so the performance and safety criteria described in Chapter 4 and Chapter 15 are met. [*The mechanical design and physical arrangement of the reactor components, together with the corrective actions of the reactor control, protection, and emergency cooling systems (when applicable) are designed to achieve these criteria, referred to as Principal Design Requirements:*

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

- *Fuel damage, defined as penetration of the fuel cladding, is predicted not to occur during normal operation and anticipated operational transients.*

- *Materials used in the fuel assembly and in-core control components are selected to be compatible in a pressurized water reactor environment.*

- *For normal operation and anticipated transient conditions, the minimum DNBR calculated using the WRB-2M correlation is greater than or equal to 1.14.*

- *Fuel melting will not occur at the overpower limit for Condition I or II events.*

- *The maximum fuel rod cladding temperature following a loss-of-coolant accident is calculated to be less than 2200°F.*

- *For normal operation and anticipated transient conditions, the calculated core average linear power, including densification effects, is less than or equal to 5.718 kw/ft for the initial fuel cycle.*

- *For normal operation and anticipated transient conditions, the calculated total heat flux hot channel factor, F_Q, is less than or equal to 2.60 for the initial fuel cycle.*

- *Calculated rod worths provide sufficient reactivity to account for the power defect from full power to zero power and provide the required shutdown margin, with allowance for the worst stuck rod.*

- *Calculations of the accidental withdrawal of two control banks using the maximum reactivity change rate predict that the peak linear heat rate and DNBR limits are met.*

- *The maximum rod control cluster assembly and gray rod speed (or travel rate) is 45 inches per minute.*

- *The control rod drive mechanisms are hydrotested after manufacture at a minimum of 125 percent of system design pressure.*

- *For the initial fuel cycle, the fuel rod temperature coefficient is calculated to be negative for power operating conditions.*

- *For the initial fuel cycle, the moderator temperature coefficient is calculated to be negative for power operating conditions.*]*

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

4.1.2 Combined License Information

This section contains no requirement for additional information to be provided in support of Combined License.

4.1.3 References

1. Letter from N. J. Liparulo (Westinghouse) to J. E. Lyons (NRC), "Transmittal of Response to NRC Request for Information on Wolf Creek Fuel Design Modifications," NSD-NRC-97-5189, June 30, 1997.

2. Letter from N. J. Liparulo (Westinghouse) to R. C. Jones (NRC), "Transmittal of Presentation Material for NRC/Westinghouse Fuel Design Change Meeting on April 15, 1996," NSD-NRC-96-4964, April 22, 1996.

3. Letter from Westinghouse to NRC, "Fuel Criteria Evaluation Process Notification for the 17x17 Robust Fuel Assembly with IFM Grid Design," NSD-NRC-98-5796, October 13, 1998.

4. Letter from H. A. Sepp (Westinghouse) to T. E. Collins (NRC), "Notification of FCEP Application for WRB-1 and WRB-2 Applicability to the 17x17 Modified LPD Grid Design for Robust Fuel Assembly Application," NSD-NRC-98-5618, March 25, 1998.

5. Letter from H. A. Sepp (Westinghouse) to T. E. Collins (NRC), "Fuel Criteria Evaluation Process Notification for the Revised Guide Thimble Dashpot Design for the 17x17 XL Robust Fuel Assembly Design," NSD-NRC-98-5722, June 23, 1998.

6. Davidson, S. L., and Kramer, W. R., (Ed.), "Reference Core Report Vantage 5 Fuel Assembly," WCAP-10444-P-A (Proprietary), September 1985 and WCAP-10445-A (Non-Proprietary), December 1983.

7. Davidson, S. L., (Ed.), "VANTAGE 5H Fuel Assembly," Addendum 2-A, WCAP-10444-P-A (Proprietary) and WCAP-10445-NP-A (Non-Proprietary), February 1989.

8. Davidson, S. L., and Nuhfer, D. L., (Ed.), "VANTAGE+ Fuel Assembly Reference Core Report," WCAP-12610-P-A (Proprietary) and WCAP-14342-A (Non-Proprietary), April 1995.

9. [*Davidson, S. L. (Ed.), "Fuel Criteria Evaluation Process," WCAP-12488-A (Proprietary) and WCAP-14204-A (Non-Proprietary), October 1994.*]*

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

10. Letter, Peralta, J. D. (NRC) to Maurer, B. F. (Westinghouse), "Approval for Increase in Licensing Burnup Limit to 62,000 MWD/MTU (TAC No. MD1486)," May 25, 2006.

11. Conner, M. E., et al. "Enhanced GRCA Rodlet Design," WCAP-16943-P-A (Proprietary) and WCAP-16943-NP-A (Non-Proprietary), September 2012.

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

Table 4.1-1 (Sheet 1 of 4)

REACTOR DESIGN COMPARISON TABLE

Thermal and Hydraulic Design Parameters	AP1000	AP600	Typical XL Plant
Reactor core heat output (MWt)	3400	1933	3800
Reactor core heat output (10^6 Btu/hr)	11,601	6596	12,969
Heat generated in fuel (%)	97.4	97.4	97.4
System pressure, nominal (psia)	2250	2250	2250
System pressure, minimum steady-state (psia)	2190	2200	2204
Minimum departure from nuclear boiling (DNBR) for design transients Typical flow channel Thimble (cold wall) flow channel	 >1.25[d] >1.25[d]	 >1.23 >1.22	 >1.26 >1.24
Departure from nucleate boiling (DNB) correlation[b]	WRB-2M[b]	WRB-2	WRB-1[a]
Coolant Flow[c]			
Total vessel thermal design flow rate (10^6 lbm/hr) Effective flow rate for heat transfer (10^6 lbm/hr) Effective flow area for heat transfer (ft^2) Average velocity along fuel rods (ft/s) Average mass velocity (10^6 lbm/hr-ft^2)	113.5 106.8 41.8 15.8 2.55	72.9 66.3 38.5 10.6 1.72	145.0 132.7 51.1 16.6 2.60
Coolant Temperature[c][e]			
Nominal inlet (°F) Average rise in vessel (°F) Average rise in core (°F) Average in core (°F) Average in vessel (°F)	535.0 77.2 81.4 578.1 573.6	532.8 69.6 75.8 572.6 567.6	561.2 63.6 68.7 597.8 593.0
Heat Transfer			
Active heat transfer surface area (ft^2) Avg. heat flux (BTU/hr-ft^2) Maximum heat flux for normal operation (BTU/hr-ft^2)[f] Average linear power (kW/ft)[g] Peak linear power for normal operation (kW/ft)[f][g] Peak linear power (kW/ft)[f][h] (Resulting from overpower transients/operator errors, assuming a maximum overpower of 118%)	56,700 199,300 518,200 5.72 14.9 ≤22.45	44,884 143,000 372,226 4.11 10.7 22.5	69,700 181,200 489,200 5.20 14.0 ≤22.45

Table 4.1-1 (Sheet 2 of 4)

REACTOR DESIGN COMPARISON TABLE

Thermal and Hydraulic Design Parameters	AP1000	AP600	Typical XL Plant
Heat flux hot channel factor (F_Q)	2.60	2.60	2.70
Peak fuel center line temperature (°F) (For prevention of center-line melt)	4700	4700	4700
Fuel assembly design	17x17 XL Robust Fuel	17x17	17x17 XL Robust Fuel/ No IFM
Number of fuel assemblies	157	145	193
Uranium dioxide rods per assembly	264	264	264
Rod pitch (in.)	0.496	0.496	0.496
Overall dimensions (in.)	8.426 x 8.426	8.426 x 8.426	8.426 x 8.426
Fuel weight, as uranium dioxide (lb)	211,588	167,360	261,000
Clad weight (lb)	43,105	35,555	63,200
Number of grids per assembly Top and bottom - (Ni-Cr-Fe Alloy 718) Intermediate Intermediate flow mixing Protective Grid - (Ni-Cr-Fe Alloy 718)	 2[i] 8 ZIRLO 4 ZIRLO 1	 2[i] 7 Zircaloy-4 or 7 ZIRLO 4 Zircaloy-4 or 5 ZIRLO 1	 2 8 ZIRLO 0 1
Loading technique, first cycle	5 region nonuniform	3 region nonuniform	3 region nonuniform
Fuel Rods			
Number	41,448	38,280	50,952
Outside diameter (in.)	0.374	0.374	0.374
Diametral gap (non-IFBA) (in.)	0.0065	0.0065	0.0065
Clad thickness (in.)	0.0225	0.0225	0.0225
Clad material	ZIRLO	Zircaloy-4 or ZIRLO	Zircaloy-4/ ZIRLO
Fuel Pellets			
Material	UO_2 sintered	UO_2 sintered	UO_2 sintered
Density (% of theoretical)	95.5	95	95
Diameter (in.)	0.3225	0.3225	0.3225
Length (in.)	0.387	0.387	0.387

Table 4.1-1 (Sheet 3 of 4)

REACTOR DESIGN COMPARISON TABLE

Rod Cluster Control Assemblies	AP1000	AP600	Typical XL Plant
Neutron Absorber			
RCCA GRCA	24 Ag-In-Cd rodlets 24 Tungsten rodlets	24 Ag-In-Cd rodlets 20 304 SS rodlets 4 Ag-In-Cd rodlets	24 Hafnium or Ag-In-Cd
Cladding material	Type 304 or 304L SS, cold-worked	Type 304 SS, cold-worked	Type 304 SS, cold-worked
Clad thickness (in.) - RCCA - GRCA	0.0185 0.0255	0.0185 0.0185	0.0185
Number of clusters	53 RCCAs 16 GRCAs	45 RCCAs 16 GRCAs	57 RCCAs 0 GRCAs
Core Structure			
Core barrel, ID/OD (in.)	133.75/137.75	133.75/137.75	148.0/152.5
Thermal shield	Neutron Panel	None	Neutron Panel
Baffle thickness (in.)	Core Shroud	Radial reflector	0.875
Structure Characteristics			
Core diameter, equivalent (in.)	119.7	115.0	132.7
Core height, cold, active fuel (in.)	168.0	144.0	168.0

Table 4.1-1 (Sheet 4 of 4)

REACTOR DESIGN COMPARISON TABLE

Fuel Enrichment First Cycle (Weight Percent)	**AP1000[j]**	**AP600**	**Typical XL Plant**
Region 1	0.74 / --	1.90	Typical
Region 2	1.58 / --	2.80	3.8 to 4.4
Region 3	3.20 / 1.58	3.70	(5.0 Max)
Region 4	3.776 / 3.20		
Region 5	4.376 / 3.20		

Notes:

a. WRB-2M will be used in future reloads.
b. See subsection 4.4.2.2.1 for the use of the ABB-NV, WLOP, WRB-2 and WRB-2M correlations.
c. Flow rates and temperatures are based on 10 percent steam generator tube plugging for the AP600 and AP1000 designs.
d. The Design Limit DNBR is 1.25.
e. Coolant temperatures based on thermal design flow (for AP600 and AP1000).
f. Based on F_Q of 2.60 for AP600 and AP1000.
g. Based on densified active fuel length. The value for AP1000 is rounded to 5.72 kW/ft.
h. See subsection 4.3.2.2.6.
i. The top grid will be fabricated of nickel-chromium-iron Alloy 718.
j. For the AP1000 design, the assembly average enrichments are given for the mid-zone and axial blanket regions.

Table 4.1-2 (Sheet 1 of 2)

ANALYTICAL TECHNIQUES IN CORE DESIGN

Analysis	Technique	Computer Code	Subsection Referenced
Mechanical design of core internals loads, deflections, and stress analysis	Static and dynamic modeling	BLOWDOWN code, FORCE, finite element structural analysis code, and others	3.7.2.1 3.9.2 3.9.3
Fuel rod design Fuel performance characteristics (such as, temperature, internal pressure, and clad stress)	Semi-empirical thermal model of fuel rod with considerations such as fuel density changes, heat transfer, and fission gas release.	Westinghouse fuel rod design model	4.2.1.1 4.2.3.2 4.2.3.3 4.3.3.1 4.4.2.11
Nuclear design Cross-sections and group constants	Microscopic data; macroscopic constants for homogenized core regions	Modified ENDF/B library with PHOENIX-P or PARAGON	4.3.3.2 4.3.3.3
X-Y and X-Y-Z power distributions, fuel depletion, critical boron concentrations, X-Y and X-Y-Z xenon distributions, reactivity coefficients	2-group diffusion theory, 2-group nodal theory	ANC (2-D or 3-D)	4.3.3.3
Axial power distributions, control rod worths, and axial xenon distribution	1-D, 2-group diffusion theory	APOLLO	4.3.3.3
Fuel rod power	Integral transport theory	LASER	4.3.3.1
Effective resonance temperature	Monte Carlo weighing function	REPAD	4.3.3.1
Criticality of reactor and fuel assemblies	3-D, Monte Carlo theory	MCNP4a	4.3.2.6
Vessel irradiation	Multigroup spatial dependent transport theory	DOT	4.3.2.8
Thermal-hydraulic design steady state	Subchannel analysis of local fluid conditions in rod bundles, including inertial and cross-flow resistance terms; solution progresses from core-wide to hot assembly to hot channel.	VIPRE-01	4.4.4.5.2

Table 4.1-2 (Sheet 2 of 2)

ANALYTICAL TECHNIQUES IN CORE DESIGN

Analysis	Technique	Computer Code	Subsection Referenced
Transient departure from nucleate boiling	Subchannel analysis of local fluid conditions in rod bundles during transients by including accumulation terms in conservation equations; solution progresses from core-wide to hot assembly to hot channel.	VIPRE-01	4.4.4.5.4

Table 4.1-3 **DESIGN LOADING CONDITIONS FOR REACTOR CORE COMPONENTS**
• Fuel assembly weight and core component weights (burnable absorbers, sources, RCCA, GRCA, thimble plug)
• Fuel assembly holddown spring forces and core component spring forces
• Internals weight
• Control rod trip (equivalent static load)
• Differential pressure
• Spring preloads
• Coolant flow forces (static)
• Temperature gradients
• Thermal expansion
• Interference between components
• Vibration (mechanically or hydraulically induced)
• Operational transients listed in Table 3.9-1
• Pump overspeed
• Seismic loads (safe shutdown earthquake)
• Blowdown forces (due to pipe rupture)

4.2 Fuel System Design

The plant conditions for design are divided into four categories.

- Condition I - normal operation and operational transients
- Condition II - events of moderate frequency
- Condition III - infrequent incidents
- Condition IV - limiting faults

Chapter 15 describes bases and plant operation and events involving each condition.

The reactor is designed so that its components meet the following performance and safety criteria:

- The mechanical design and physical arrangement of the reactor core components, together with corrective actions of the reactor control, protection, and emergency cooling systems (when applicable) provide that:

  – Fuel damage, that is, breach of fuel rod clad pressure boundary, is not expected during Condition I and Condition II events. A very small amount of fuel damage may occur. This is within the capability of the plant cleanup system and is consistent with the plant design bases.

  – The reactor can be brought to a safe state following a Condition III event with only a small fraction of fuel rods damaged. The fraction of fuel rods damaged must be limited to meet the dose guidelines identified in Chapter 15 although sufficient fuel damage might occur to preclude immediate resumption of operation.

  – The reactor can be brought to a safe state and the core kept subcritical with acceptable heat transfer geometry following transients arising from Condition IV events.

- The fuel assemblies are designed to withstand non-operational loads induced during shipping, handling, and core loading without exceeding the criteria of subsection 4.2.1.5.1.

- The fuel assemblies are designed to accept control rod insertions to provide the required reactivity control for power operations and reactivity shutdown conditions.

- The fuel assemblies have provisions for the insertion of in-core instrumentation.

- The reactor vessel and internals, in conjunction with the fuel assembly structure, directs reactor coolant through the core. Because of the resulting flow distribution and bypass flow, the heat transfer performance requirements are met for the modes of operation.

The following subsection provides the fuel system design bases and design limits. It is consistent with the criteria of the Standard Review Plan, Section 4.2.

Consistent with the growth in technology, Westinghouse modifies fuel system designs. These modifications utilize NRC approved methods. [*A set of design fuel criteria to be satisfied by new fuel designs was issued to the NRC in WCAP-12488-A and its addendums (Reference 1)*]* and also presented below in subsection 4.2.1.

4.2.1 Design Basis

The fuel rod and fuel assembly design bases are established to satisfy the general performance and safety criteria presented in Section 4.2 of the Standard Review Plan. [*The design bases and acceptance limits used by Westinghouse are also described in the Westinghouse Fuel Criteria Evaluation Process, WCAP-12488-A and its addendums (Reference 1).*]*

The fuel rods are designed to satisfy the fuel rod design criteria for rod burnup levels up to the design discharge burnup using the extended burnup design methods described in the Extended Burnup Evaluation report, WCAP-10125-P-A (Reference 2).

The AP1000 fuel rod design considers effects such as fuel density changes, fission gas release, clad creep, and other physical properties which vary with burnup. The integrity of the fuel rods is provided by designing to prevent excessive fuel temperatures as discussed in subsection 4.2.1.2.1; excessive internal rod gas pressures due to fission gas releases as discussed in subsections 4.2.1.3.1 and 4.2.1.3.2; and excessive cladding stresses, strains, and strain fatigue, as discussed in subsections 4.2.1.1.2 and 4.2.1.1.3. The fuel rods are designed so that the conservative design bases of the following events envelope the lifetime operating conditions of the fuel. For each design basis, the performance of the limiting fuel rod, with appropriate consideration for uncertainties, does not exceed the limits specified by the design basis. The detailed fuel rod design also establishes such parameters as pellet size and density, clad/pellet diametral gap, gas plenum size, and helium pre-pressurization level.

Integrity of the fuel assembly structure is provided by setting limits on stresses and deformations due to various loads and by preventing the assembly structure from interfering with the functioning of other components. Three types of loads are considered:

- Non-operational loads, such as those due to shipping and handling
- Normal and abnormal loads, which are defined for Conditions I and II
- Abnormal loads, which are defined for Conditions III and IV

The design bases for the in-core control components are described in subsection 4.2.1.6.

4.2.1.1 Cladding

4.2.1.1.1 Mechanical Properties

The ZIRLO cladding material combines neutron economy (low absorption cross-section); high corrosion resistance to coolant, fuel, and fission products; and high strength and ductility at operating temperatures. ZIRLO is an advanced zirconium based alloy that has the same or similar properties and advantages as Zircaloy-4 and was developed to support extended fuel burnup. WCAP-12610-P-A (Reference 5) provides a discussion of chemical and mechanical properties of the ZIRLO cladding material and a comparison to Zircaloy-4.

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

4.2.1.1.2 Stress-Strain Limits

Clad Stress

[*The volume average effective stress calculated with the Von Mises equation (considering interference due to uniform cylindrical pellet-clad contact, caused by pellet thermal expansion, pellet swelling and uniform clad creep, and pressure differences) is less than the 0.2 percent offset yield stress with due consideration to temperature and irradiation effects for Condition I and II events, WCAP-12488-A (Reference 1).*]* While the clad has some capability for accommodating plastic strain, the yield stress has been accepted as a conservative design limit. The allowable stress limits due to Condition III and IV loadings, described in subsection 4.2.1.5.3, are also applied to the fuel rod.

Clad Strain

[*The total plastic tensile creep strain due to uniform clad creep, and uniform cylindrical fuel pellet expansion associated with fuel swelling and thermal expansion is less than one percent from the unirradiated condition, WCAP-12488-A (Reference 1).*]* The acceptance limit for fuel rod clad strain during Condition II events is that the total tensile strain due to uniform cylindrical pellet thermal expansion is less than one percent from the pre-transient value. These limits are consistent with proven practice.

4.2.1.1.3 Fatigue and Vibration

Fatigue

[*The usage factor due to cycle fatigue is less than 1.0, WCAP-12488-A (Reference 1).*]* That is, for a given strain range, the number of strain fatigue cycles are less than those required for failure. The fatigue curve is based on a safety factor of two on the stress amplitude or a safety factor of 20 on the number of cycles, whichever is more conservative.

Vibration

Potential fretting wear due to vibration is prevented, giving confidence that the stress-strain limits are not exceeded during design life. Fretting of the clad surface can occur due to flow-induced vibration between the fuel rods and fuel assembly grid springs. Vibration and fretting forces may vary during the fuel life due to clad diameter creep down combined with grid spring relaxation.

4.2.1.1.4 Chemical Properties

Chemical properties of the ZIRLO cladding are discussed in WCAP-12610 (Reference 5).

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

4.2.1.2 Fuel Material

4.2.1.2.1 Thermal-Physical Properties

The center temperature of the hottest pellet is below the melting temperature of the uranium dioxide. The melting temperature of unirradiated uranium dioxide, 5080°F, decreases by 58°F per 10,000 megawatt days per metric ton of uranium, as discussed in WCAP-9179 (Reference 4). Fuel melting will not occur at the overpower limit for Condition I or II events. This provides sufficient margin for uncertainties as described in subsection 4.4.2.9.

The nominal design density of the fuel is approximately 95.5 percent of the theoretical density. Additional information on fuel properties is provided in WCAP-9179 (Reference 4).

4.2.1.2.2 Fuel Densification and Fission Product Swelling

The design bases and models used for fuel densification and swelling are provided in WCAP-10851-P-A (Reference 7), and WCAP-15063-P-A, Revision 1 (Reference 21).

4.2.1.2.3 Chemical Properties

WCAP-9179 (Reference 4) and WCAP-12610 (Reference 5) provide the basis for justifying that no adverse chemical interactions occur between the fuel and its adjacent material.

4.2.1.3 Fuel Rod Performance

4.2.1.3.1 Fuel Rod Models

The basic fuel rod models and the ability to predict fuel rod operating characteristics are given in WCAP-15063-P-A, Revision 1 (Reference 21) and subsection 4.2.3.

4.2.1.3.2 Mechanical Design Limits

Cladding collapse is precluded during the fuel rod design lifetime. Current generation Westinghouse fuel is sufficiently stable with respect to fuel densification. Significant axial gaps in the pellet stack necessary for clad flattening do not occur and therefore, clad flattening will not occur. Clad flattening methodologies are described in WCAP-13589-A, (Reference 8) and WCAP-8377 (Reference 22).

The rod internal gas pressure remains below the value which causes the fuel/clad diametral gap to increase due to outward cladding creep during steady-state operation. Rod pressure is also limited such that extensive departure from nucleate boiling propagation does not occur as discussed in WCAP-8963-P-A (Reference 9).

4.2.1.4 Spacer Grids

4.2.1.4.1 Mechanical Limits and Materials Properties

The grid component strength criteria are based on experimental tests. The limit is established at the 95-percent confidence level on the true mean crush strength at operating temperature. This limit is sufficient to provide that, under worst-case combined seismic and pipe rupture event, the core will maintain a geometry amenable to cooling. As an integral part of the fuel

assembly structure, the grids satisfy the applicable fuel assembly design bases and limits defined in subsection 4.2.1.5.

The grid material and chemical properties are given in WCAP-9179 (Reference 4) and WCAP-12610 (Reference 5).

4.2.1.4.2 Vibration and Fatigue

The grids provide sufficient fuel rod support to limit fuel rod vibration and maintain clad fretting wear within acceptable limits (defined in subsection 4.2.1.1).

4.2.1.5 Fuel Assembly Structural Design

As discussed in subsection 4.2.1, the structural integrity of the fuel assemblies is provided by setting design limits on stresses and deformations due to various non-operational, operational, and accident loads. These limits are applied to the design and evaluation of the top and bottom nozzles, guide thimbles, grids, and thimble joints. [*Design changes to the fuel assembly structure qualify for evaluation in WCAP-12488-A (Reference 1).*]*

The design bases for evaluating the structural integrity of the fuel assemblies are discussed in subsections 4.2.1.5.1 through 4.2.1.5.3.

4.2.1.5.1 Non-Operational

The non-operational load is a loading of 4 g axial (longitudinal) and 6 g lateral (transverse) with dimensional stability.

4.2.1.5.2 Normal Operation and Operational Transients (Condition I) and Events of Moderate Frequency (Condition II)

For the normal operation (Condition I) and upset (Condition II) conditions, the fuel assembly component structural design criteria are established for the two primary material categories, austenitic steels and zirconium alloys. The stress categories and strength theory presented in the ASME Code, Section III, are used as a general guide. The maximum shear theory (Tresca criterion) for combined stresses is used to determine the stress intensities for the austenitic steel components. The stress intensity is defined as the largest numerical difference between the various principal stresses in a three-dimensional field. The design stress intensity value, S_m, for austenitic steels and zirconium alloys is given by the lowest of the following:

- One-third of the specified minimum tensile strength or two-thirds of the specified minimum yield strength at room temperature
- One-third of the tensile strength or 90 percent of the yield strength at temperature, but not to exceed two-thirds of the specified minimum yield strength at room temperature

The stress limits for the austenitic steel components are given below. Stress nomenclature follows the ASME Code, Section III.

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

Stress Intensity Limits

Categories	Limit
General primary membrane stress intensity	S_m
Local primary membrane stress intensity	$1.5\ S_m$
Primary membrane plus bending stress intensity	$1.5\ S_m$
Total primary plus secondary stress intensity	$3.0\ S_m$

The zirconium alloy structural components, which consist of guide thimbles and fuel tubes, are in turn subdivided into two categories because of material difference and functional requirements. The fuel tube design criteria are covered separately in subsection 4.2.1.1. The maximum shear theory is used to evaluate the guide thimble design. For conservative purposes, the zirconium alloy unirradiated properties are used to define the stress limits.

4.2.1.5.3 Infrequent Incidents (Condition III) and Limiting Faults (Condition IV)

Typical worse case abnormal loads during Conditions III and IV are represented by seismic and pipe rupture loadings. The design criteria for this category of loadings are as follows:

- Deflections or excessive deformation of components cannot interfere with capability of insertion of the control rods or emergency cooling of the fuel rods.

- The fuel assembly structural components stresses under faulted conditions are evaluated primarily using the methods outlined in Appendix F of the ASME Code, Section III. Since the current analytical methods use linear elastic analysis, the stress allowables are defined as the smaller value of $2.4\ S_m$ or $0.70\ S_u$ for primary membrane and $3.6\ S_m$ or $1.05\ S_u$ for primary membrane plus primary bending. For the austenitic steel fuel assembly components, the stress intensity is defined in accordance with the rules described in the previous section for normal operating conditions. For the zirconium alloy components, the stress intensity limits are set at two-thirds of the material yield strength, S_y, at reactor operating temperature. This results in zirconium alloy stress limits being the smaller value of $1.6\ S_y$ or $0.70\ S_u$ for primary membrane and $2.4\ S_y$ or $1.05\ S_u$ for primary membrane plus bending.

- For conservative purposes, the zirconium alloy unirradiated properties are used to define the stress limits.

The material and chemical properties of the fuel assembly components are given in WCAP-9179 (Reference 4) and WCAP-12610 (Reference 5). Subsection 4.2.3.4 discusses the spacer grid crush testing.

Thermal-hydraulic design is discussed in Section 4.4.

4.2.1.6 In-core Control Components

The in-core control components are subdivided into permanent and temporary devices. The permanent components are the rod cluster control assemblies, gray rod cluster assemblies, thimble plugs, and secondary neutron source assemblies. The temporary components are the primary neutron source assemblies (which are normally used only in the initial core) and the burnable absorber assemblies. For some reloads, the use of burnable absorbers may be necessary for power distribution control and/or to achieve an acceptable moderator temperature coefficient throughout core life (See Subsection 4.3.1.2.2). [*Design changes to the in-core control components qualify for evaluation using the criteria defined in WCAP-12488-A (Reference 1).*]*

Materials are selected for:

- Compatibility in a pressurized water reactor environment
- Adequate mechanical properties at room and operating temperatures
- Resistance to adverse property changes in a radioactive environment

Compatibility with interfacing components

Material properties are given in WCAP-9179 (Reference 4).

The design bases for the in-core control components are given in subsections 4.2.1.6.1 through 4.2.1.6.3.

4.2.1.6.1 Control Rods

For Conditions I and II, the stress categories and strength theory presented in the ASME Code, Section III, are used as a general guide in the design of the RCCA and GRCA structural parts in addition to absorber cladding.

Design conditions considered under the ASME Code, Section III, are as follows:

- External pressure equal to the reactor coolant system operating pressure with appropriate allowance for overpressure transients

- Wear allowance equivalent to 1000 reactor trips

- Bending of the rod due to a misalignment in the guide thimble

- Forces imposed on the rods during rod drop

- Loads imposed by the accelerations of the control rod drive mechanism

- Radiation exposure during maximum core life. The RCCA absorber material temperature does not exceed its melting temperature (1454°F for silver-indium-cadmium [Ag-In-Cd]), (see WCAP-9179, Reference 4). Similarly, the GRCA absorber material temperature does not exceed its melting temperature (6116°F for tungsten), (see WCAP-16943-P-A, Reference 25).

- Temperature effects at operating conditions

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

4.2.1.6.2 Burnable Absorber Rods

For Conditions I and II, the stress categories and strength theory presented in the ASME Code, Section III, are used as a general guide in the design of the burnable absorber cladding. For abnormal loads during Conditions III and IV, code stresses are not considered limiting. Failures of the burnable absorber rods during these conditions must not interfere with reactor shutdown or emergency cooling of the fuel rods. The burnable absorber material is nonstructural. The structural elements of the burnable absorber rod are designed to maintain the absorber geometry even if the absorber material is fractured.

To reduce the dissolved boron requirement for control of excess reactivity, wet annular burnable absorber (WABA) rods have been incorporated in the first core design. The burnable absorber material is boron carbide contained in an alumina matrix. Thermal-physical and gas release properties of alumina-boron carbide are described in WCAP-9179 (Reference 4) and WCAP-10021-P-A (Reference 10). Discrete burnable absorber rods are designed so that the absorber temperature does not exceed 1200°F during normal operation or an overpower transient. The 1200°F maximum temperature helium gas release in a discrete burnable absorber rod will not exceed 30 percent of theoretical. See WCAP-10021-P-A (Reference 10).

4.2.1.6.3 Neutron Source Rods

The neutron source rods are designed to withstand the following:

- The external pressure equal to reactor coolant system operating pressure with appropriate allowance for overpressure transients
- An internal pressure equal to the pressure generated by released gases over the source rod life

4.2.1.7 Surveillance Program

Subsection 4.2.4.6 discusses the testing and fuel surveillance operation experience program that has been and is being conducted to verify the adequacy of the fuel performance and design bases. Fuel surveillance and testing results, as they become available, are used to improve fuel rod design and manufacturing processes and to confirm that the design bases and safety criteria are satisfied.

4.2.2 Description and Design Drawings

The fuel assembly, fuel rod, and in-core control component design data is given in Table 4.3-1.

Each fuel assembly consists of 264 fuel rods, 24 guide thimbles, and 1 instrumentation tube arranged within a supporting structure. The instrumentation thimble is located in the center position and provides a channel for insertion of an in-core neutron detector, if the fuel assembly is located in an instrumented core position. The guide thimbles provide channels for insertion of either a rod cluster control assembly, a gray rod cluster assembly, a neutron source assembly, a burnable absorber assembly, or a thimble plug, depending on the position

of the particular fuel assembly in the core. Figure 4.2-1 shows a cross-section of the fuel assembly array, and Figure 4.2-2 shows a fuel assembly full-length view.

The fuel rods are loaded into the fuel assembly structure so that there is clearance between the fuel rod ends and the top and bottom nozzles. The fuel rods are supported within the fuel assembly structure by fourteen grids (top grid (1), bottom grid (1), intermediate grids (8) and intermediate flow mixer (IFM) grids (4)), plus one protective grid. The top grid is fabricated from nickel-chromium-iron Alloy 718. The bottom grid is fabricated from nickel-chromium-iron Alloy 718. The intermediate grids and the IFM grids are fabricated from ZIRLO (see WCAP-12610-P-A, Reference 5). Top, bottom, and intermediate grids provide axial and lateral support to the fuel rods. In addition, the four IFM grids located near the center of the fuel assembly and between the intermediate grids provide additional fuel rod restraint. The protective grid, in combination with the debris filter bottom nozzle (DFBN), the protective zirconium oxide coated fuel cladding, and the long, solid fuel rod bottom end plug, provide debris failure mitigation.

Fuel assemblies are installed vertically in the reactor vessel and stand upright on the lower core plate, which is fitted with alignment pins to locate and orient each assembly. After the fuel assemblies are set in place, the upper support structure is installed. Alignment pins, built into the upper core plate, engage and locate the upper ends of the fuel assemblies. The upper core plate then bears down against the hold-down springs on the top nozzle of each fuel assembly to hold the fuel assemblies in place.

Improper orientation of fuel assemblies within the core is prevented by the use of an indexing hole in one corner of the top nozzle top plate. The assembly is oriented with respect to the handling tool and the core by means of a pin inserted into this indexing hole. Visual confirmation of proper orientation is also provided by an engraved identification number on the opposite corner clamp.

4.2.2.1 Fuel Rods

The fuel rods consist of uranium dioxide ceramic pellets contained in cold-worked and stress relieved ZIRLO tubing, which is plugged and seal-welded at the ends to encapsulate the fuel. ZIRLO is an advanced zirconium based alloy selected for its mechanical properties and low neutron absorption cross-section (see WCAP-12610-P-A, Reference 5). Figure 4.2-3 shows a schematic of the fuel rod. The fuel pellets are right circular cylinders consisting of slightly enriched uranium dioxide powder which has been compacted by cold pressing and then sintered to the required density. The ends of each pellet are dished slightly, to allow greater axial expansion at the pellet centerline and to increase the void volume for fission gas release. The ends of each pellet also have a small chamfer at the outer cylindrical surface which improves manufacturability, and mitigates potential pellet damage due to fuel rod handling.

Void volume and clearances are provided within the rods to accommodate fission gases released from the fuel, differential thermal expansion between the clad and the fuel, and fuel density changes during irradiation. To facilitate the extended burnup capability necessitated by longer operating cycles, the fuel rod is designed with two plenums (upper and lower) to accommodate the additional fission gas release. The upper plenum volume is maintained by a fuel pellet hold-down spring. The lower plenum volume is maintained by a standoff assembly.

Shifting of the fuel within the clad during handling or shipping, prior to core loading, is prevented by a stainless steel helical spring which bears on top of the fuel pellet stack. Assembly consists of plugging and welding the bottom of the cladding, installing the bottom plenum spacer assembly, fuel pellets and top plenum spring, and then plugging and welding the top of the rod. The solid bottom end plug has an internal grip feature and tapered end to facilitate fuel rod loading during fuel assembly fabrication and reconstitution. Additionally, the bottom end plug is designed to be sufficiently long to extend through the protective grid. The bottom section of the fuel rod has a protective zirconium oxide coated surface feature. Use of the protective grid with a longer end plug and the debris filter bottom nozzle, in addition to the coated cladding surface, constitutes a three-level debris protection package, which enhances the fuel reliability performance against trapped debris. This precludes any breach in the fuel rod pressure boundary due to clad fretting wear induced by debris trapped at the bottom section of the fuel assembly.

The fuel rods are internally pressurized with helium during the welding process to minimize compressive clad stresses and prevent clad flattening under reactor coolant operating pressures. The fuel rods are pre-pressurized and designed so that:

- The internal gas pressure mechanical design limit referred to in subsection 4.2.1.3 is not exceeded
- The cladding stress-strain limits (subsection 4.2.1.1) are not exceeded for Condition I and II events
- Clad flattening will not occur during the fuel core life

The AP1000 fuel rod design may also include axial blankets. The axial blankets consist of fuel pellets of a reduced enrichment at each end of the fuel rod pellet stack. Axial blankets reduce neutron leakage axially and improve fuel utilization. The axial blankets use chamfered pellets that are longer than the enriched pellets to help prevent accidental mixing during manufacturing. Furthermore, axial blankets have no impact on the source range detector response, since the reduction in power from the axial blanket is limited to the top and bottom 0.67 feet of the core, while the source range detectors are centered typically about three feet from the bottom of the core.

The AP1000 fuel rod design may also include annular fuel pellets in the top and bottom 8 inches of the fuel stack. These pellets can be either fully enriched or partially enriched. The annular fuel pellets provide additional void volume in the fuel rod to accommodate fission gas release.

The AP1000 fuel rods include integral fuel burnable absorbers. The integral fuel burnable absorbers are boride-coated fuel pellets. The boride-coated fuel pellets are identical to the enriched uranium dioxide pellets except for the addition of a thin boride coating less than 0.001 inch in thickness on the pellet cylindrical surface. Coated pellets occupy the central portion of the fuel column. The number and pattern of integral fuel burnable absorber rods within an assembly may vary depending on specific application. See WCAP-12610-P-A (Reference 5).

4.2.2.2 Fuel Assembly Structure

As shown in Figure 4.2-2, the fuel assembly structure consists of a bottom nozzle, top nozzle, fuel rods, guide thimbles, and grids.

4.2.2.2.1 Bottom Nozzle

The bottom nozzle serves as the bottom structural element of the fuel assembly and directs the coolant flow distribution to the assembly. The nozzle is fabricated from Type 304 and Grade CF-3 stainless steel and consists of a perforated plate, and casting which incorporates a skirt and four angle legs with bearing pads. Figure 4.2-2 illustrates this concept. The legs and skirt form a plenum to direct the inlet coolant flow to the fuel assembly. The perforated plate also prevents accidental downward ejection of the fuel rods from the fuel assembly. The bottom nozzle is fastened to the fuel assembly guide thimbles by locked thimble screws, which penetrate through the nozzle and engage with a threaded plug in each guide thimble.

Coolant flows from the plenum in the bottom nozzle, upward through the penetrations in the plate, to the channels between the fuel rods. The penetrations in the plate are positioned between the rows of the fuel rods.

In addition to serving as the bottom structural element of the fuel assembly, the bottom nozzle also functions as a debris filter. The bottom nozzle perforated plate contains a multiplicity of flow holes which are sized to minimize passage of detrimental debris particles into the active fuel region of the core while maintaining sufficient hydraulic and structural margins. Furthermore, the skirt provides improved bottom nozzle structural stability and increased design margins to reduce damage due to abnormal handling.

Axial loads (from top nozzle hold-down springs) imposed on the fuel assembly and the weight of the fuel assembly are transmitted through the bottom nozzle to the lower core plate. Indexing and positioning of the fuel assembly is controlled by alignment holes in two diagonally opposite bearing pads that mate with locating pins in the lower core plate. Lateral loads on the fuel assembly are transmitted to the lower core plate through the locating pins.

The AP1000 bottom nozzle also has a reconstitution design feature which facilitates the easy removal of the nozzle from the fuel assembly. This design incorporates a thimble screw with a circular locking cup located around the screw head. The locking cup is crimped into a local spherical radius relief on the bottom nozzle. To remove the bottom nozzle, a counterclockwise torque is applied to the thimble screw until the locking cup (detents) is relaxed and the thimble screw is removed. This reconstitutable design permits the remote unlocking, the removal, and the relocking of the thimble screws, as the same or a new bottom nozzle is reattached to the fuel assembly.

4.2.2.2.2 Top Nozzle

The reconstitutable top nozzle functions as the upper structural component of the fuel assembly and, in addition, provides a partial protective housing for the rod cluster control assembly, discrete burnable absorber, or other core components. As shown in Figure 4.2-2, the top nozzle assembly includes four sets of hold-down springs, which are secured to the top nozzle top plate. The springs are made of nickel-chromium-iron Alloy 718. The other top nozzle components are made of Type 304L and Grade CF-3 stainless steel.

The adapter plate is provided with round penetrations and slots (with semicircular ends) to permit the flow of coolant upward through the top nozzle. Other round holes are provided in the adapter plate to accept (guide thimble) inserts which are mechanically locked to the adapter plate using a lock tube. The unique design of the insert joint and lock tube are the key design features of the reconstitutable top nozzle.

The ligaments in the adapter plate cover the top of the fuel rods precluding any upward ejection of the fuel rods from the fuel assembly. The enclosure is a box-like structure which establishes the distance between the adapter plate and the top plate. The top plate has a large square hole in the center to permit access for the rod cluster control assembly, burnable absorber assembly, or other components. Hold-down springs are mounted on the top plate and are retained by retaining pins located at diagonally opposite corners of the top plate.

The top plate also contains integral pads located on the two remaining top nozzle corners. The pads include alignment holes which, when fully engaged with the reactor internals upper core plate guide pins, provide proper alignment to the fuel assembly, reactor internals, and rod control cluster assembly.

As shown in Figure 4.2-4, to remove the top nozzle assembly a tool is first inserted through a lock tube and expanded radially to engage the bottom edge of the tube. An axial force is then exerted on the tool which overrides local lock tube deformations and withdraws the lock tubes from the inserts. After the lock tubes have been removed, the nozzle assembly is removed by raising it off the upper slotted ends of the nozzle inserts, which deflect inwardly under the axial lift load.

With the top nozzle assembly removed, direct access is provided for fuel rod examination or replacement. Reconstitution is completed by the remounting of the nozzle assembly and the insertion of lock tubes. Details of this design feature, the design bases and evaluation of the reconstitutable top nozzle are given in WCAP-10444-P-A (Reference 11).

4.2.2.2.3 Guide Thimbles and Instrument Tube

The guide thimbles are structural members that provide channels for the neutron absorber rods, burnable absorber rods, neutron source rods, or other assemblies. Each guide thimble is fabricated from Zircaloy-4 or ZIRLO with constant OD and ID over the entire length. Separate dashpot tubes, which are made from Zircaloy-4 or ZIRLO tubing, are inserted into the bottom portion of the guide thimble tubes. The larger tube diameter at the top section provides a relatively large annular area necessary to permit rapid control rod insertion during a reactor trip, as well as to accommodate the flow of coolant during normal operation. Holes are provided on the guide thimble above the dashpot to reduce the rod drop time. The lower portion of the guide thimble with the dashpot tube results in a dashpot action near the end of the control rod travel during normal trip operation. The dashpot is closed at the bottom by means of an end plug, which is provided with a small flow port to avoid fluid stagnation in the dashpot volume during normal operation.

As stated previously, the AP1000 fuel assembly includes a reconstitutable top nozzle as a standard feature. To accommodate the reconstitutable feature, the top end of the zirconium alloy guide thimble is fastened to a tubular sleeve, or insert, by a three tier expansion bulge joint. An expansion tool is inserted inside the nozzle insert and guide thimble to the proper

elevation. The four lobes on the expansion tool force the guide thimble and insert outward locally to a predetermined diameter, therefore joining the two components.

Upon installation of the top nozzle assembly, the bulge near the top of the nozzle insert is captured in a corresponding groove in the hole of the top nozzle adapter plate. As shown in Figure 4.2-4, the mechanical connection between the nozzle insert-guide thimble and top nozzle is made by insertion of a lock tube into the insert. The design of the top grid sleeve-guide thimble and top nozzle insert-guide thimble bulge joint connections have been mechanically tested and found to meet applicable design criteria.

The fuel rod support grids are secured to the guide thimbles using a similar bulge joint connection to create an integral structure. Attachment of the intermediate mixing vane and intermediate flow mixer (IFM) zirconium alloy grids to the guide thimbles is performed using the fastening technique depicted in Figures 4.2-5 and 4.2-6.

The intermediate mixing vane and intermediate flow mixer grids employ a single tier bulge connection between the grid sleeve and guide thimble as compared to the two tier bulge connection used for the top grid. The design of the single tier bulge joint connection has also been mechanically tested and meets the design requirements.

The bottom nickel-chromium-iron Alloy 718 grid is secured to the guide thimble assembly by a double tier bulge connection between the grid sleeve and guide thimble. The design of the double tier bulge joint connection has also been mechanically tested and meets the design requirements.

The lower end of the guide thimble is fitted with a welded end plug. The nickel-chromium-iron Alloy 718 protective grid is secured to the guide thimble assembly by nickel-chromium-iron Alloy 718 spacers that are spot-welded to the grid. As shown in Figure 4.2-7, the spacer is captured between the guide thimble end plug and the bottom nozzle by means of a (thimble) locking screw.

The described methods of grid fastening are standard and have been used successfully since the introduction of zirconium alloy guide thimbles in 1969.

The central instrumentation tube in each fuel assembly is constrained by seating in counterbores located in both top and bottom nozzles. The instrumentation tube has a constant diameter and provides an unrestricted passageway for the in-core neutron detector which enters the fuel assembly from the top nozzle. Furthermore, the instrumentation tube is secured to the top, bottom, and mid-grids with bulge joint connections similar to those previously discussed for securing the grids to the guide thimbles.

4.2.2.2.4 Grid Assemblies

As shown in Figure 4.2-2, the fuel rods are supported at intervals along their lengths by grid assemblies which maintain the lateral spacing between the rods throughout the design life of the assembly. Each fuel rod is given support at six contact points within each grid by the combination of support dimples and springs. The grid assembly consists of individual slotted straps assembled and interlocked into an egg-crate type arrangement with the straps permanently joined at their points of intersection. The straps may contain springs, support dimples, and mixing vanes; or any such combination.

Two types of structural grid assemblies are used on the AP1000 fuel assembly. One type, with mixing vanes projecting from the edges of the straps into the coolant stream, is used in the high heat flux region of the fuel assemblies to promote mixing of the coolant. The other type, located at the top and bottom of the assembly, does not contain mixing vanes on the internal straps. The outside straps on the grids contain mixing vanes that, in addition to their mixing function, aid in guiding the grids and fuel assemblies past projecting surfaces during handling or during loading and unloading of the core.

Because of its corrosion resistance and high strength properties, the bottom grid material chosen for the AP1000 fuel assembly design is nickel-chromium-iron Alloy 718. The top grid is fabricated from nickel-chromium-iron Alloy 718. The magnitude of the grid restraining force on the fuel rod is set high enough to minimize possible fretting, without overstressing the cladding at the points of contact between the grids and fuel rods. The grid assemblies are designed to allow axial thermal expansion of the fuel rods without imposing restraint sufficient to develop buckling or distortion of the fuel rods.

The eight intermediate (mixing vane), or structural grids on the AP1000 fuel assembly are made of ZIRLO. This material was selected to take advantage of the material's inherent low neutron capture cross-section. The zirconium alloy grids have thicker straps than the nickel-chromium-iron alloy grids. The zirconium alloy grid incorporates the same grid cell support configuration as the nickel-chromium-iron alloy grid. The zirconium alloy interlocking strap joints and grid/sleeve joints are fabricated by laser welding, whereas the nickel-chromium-iron alloy grid joints (except the protective grid) are brazed. The interlocking strap joints for the protective grid are also fabricated by laser welding. The mixing vanes incorporated in the zirconium alloy intermediate grids induce additional flow mixing among the various flow channels in a fuel assembly as well as between adjacent fuel assemblies. This additional flow mixing enhances thermal performance.

As shown in Figure 4.2-2, the intermediate flow mixer grids are located at selected spans between the zirconium alloy mixing vane structural grids and incorporate a similar mixing vane array. Their prime function is mid-span flow mixing in the hotter fuel assembly spans. Each intermediate flow mixer grid cell contains four dimples that are designed to prevent mid-span channel closure in the spans containing intermediate flow mixers and fuel rod contact with the mixing vanes. This simplified cell arrangement allows short grid cells so that the intermediate flow mixer grid can accomplish its flow mixing objective with minimal pressure drop.

The intermediate flow mixer (IFM) grids, like the mixing vane grids, are fabricated from ZIRLO. The intermediate flow mixer grids are manufactured using the same basic techniques as the zirconium alloy structural grid assemblies and are joined to the guide thimbles via sleeves which are welded at the bottom of appropriate grid cells.

Grid impact testing has been performed on zirconium alloy structural grids and the intermediate flow mixer grids indicative of the AP1000 design. The purpose of the testing was to determine the dynamic buckling, or crush, strength of the grids. The grid impact testing was performed at an elevated temperature of 600°F. This temperature is a conservative value representing the core average temperature at the mid-grid locations.

The intermediate flow mixer grids are not intended to be structural members. The intermediate flow mixer grids do, however, share the loads of the structural grids during

faulted loading and, as such, contribute to enhance the load carrying capability of the AP1000 fuel assembly.

The dynamic crush strength of the AP1000 structural grids and intermediate flow mixer grids envelope the calculated grid impact loading during combined seismic and pipe rupture events. A coolable geometry is, therefore, provided at the intermediate flow mixer grid elevations, as well as at the structural grid elevations.

4.2.2.3 In-core Control Components

Reactivity control is provided by neutron absorbing rods, gray rods, burnable absorber rods, and a soluble chemical neutron absorber (boric acid). The boric acid concentration is varied to control long-term reactivity changes such as:

- Fuel depletion and fission product buildup
- Cold to hot, zero power reactivity changes
- Reactivity change produced by intermediate-term fission products such as xenon and samarium
- Burnable absorber depletion

The chemical and volume control system, which is used to adjust the level of boron in the coolant, is discussed in Section 9.3.

The rod cluster control assemblies provide reactivity control for:

- Shutdown
- Reactivity changes due to coolant temperature changes in the power range
- Reactivity changes associated with the power coefficient of reactivity
- Reactivity changes due to void formation

A negative power coefficient is maintained at hot, full-power conditions throughout the entire cycle to reduce possible deleterious effects caused by a positive coefficient during pipe rupture or loss-of-flow accidents. The first fuel cycle needs more excess reactivity than subsequent cycles due to the loading of fresh (unburned) fuel. Since soluble boron alone is insufficient to provide a negative moderator coefficient, burnable absorber assemblies are also used. Use of burnable absorber assemblies during reloads is discussed in subsection 4.3.1.2.2.

The most effective reactivity control components are the rod cluster control assemblies and the corresponding drive rod assemblies, which along with the gray rod cluster assemblies, are the only kinetic parts in the reactor. Figure 4.2-8 identifies the rod cluster control and drive rod assembly, in addition to the arrangement of these components in the reactor relative to the interfacing fuel assembly, guide thimbles, and control rod drive mechanism. The arrangement for the gray rod cluster assemblies is the same.

As shown in Figure 4.2-8, the guidance system for the rod cluster control assembly is provided by the guide thimbles. The guide thimbles provide two regimes of guidance: first,

in the lower section, a continuous guidance system provides support immediately above the core, which protects the rod against excessive deformation and wear caused by hydraulic loading. Second, the region above the continuous section provides support and guidance at uniformly spaced intervals.

As shown in Figure 4.2-9, the envelope of support is determined by the pattern of the control rod cluster. The guide thimbles provide alignment and support of the control rods, spider body, and drive rod while maintaining trip times at or below required limits.

Subsections 4.2.2.3.1 through 4.2.2.3.4 describe each reactivity control component in detail. The control rod drive mechanism assembly is described in subsection 3.9.4. The neutron source assemblies provide a means of monitoring the core during periods of low neutron activity.

4.2.2.3.1 Rod Cluster Control Assemblies

The rod cluster control assemblies are divided into two categories: control and shutdown. The control groups compensate for reactivity changes due to variations in operating conditions of the reactor, that is, power and temperature variations. Two nuclear design criteria have been employed for selection of the control group. First, the total reactivity worth must be adequate to meet the nuclear requirements of the reactor. Second, in view of the fact that these rods may be partially inserted at power operation, the total power peaking factor should be low enough to confirm that the power capability is met. The control and shutdown groups provide adequate shutdown margin.

As illustrated in Figure 4.2-9, a rod cluster control assembly is comprised of a group of individual neutron absorber rods fastened at the top end to a common spider assembly.

The absorber material used in these rods is silver-indium-cadmium alloy, which is essentially "black" to thermal neutrons and has sufficient additional resonance absorption to significantly increase worth. As such, these rods are sometimes referred to as "black" rods. As shown in Figure 4.2-10, the absorber material is in the form of solid bars sealed in cold-worked stainless steel tubes. Sufficient diametral and end clearance is provided to accommodate relative thermal expansions.

The control rods have bottom plugs with bullet-like tips to reduce the hydraulic drag during reactor trip and to guide smoothly into the dashpot section of the fuel assembly guide thimbles.

The material used in the absorber rod end plugs is Type 308 or 308L stainless steel. The design stresses used for these materials are the same as those defined in the ASME B&PV Code for Type 304 or 304L stainless steel which have essentially the same strength properties as Type 308 and 308L stainless steel, respectively.

The allowable stresses used as a function of temperature are listed in Table 2A of the ASME Code, Section II, Part D. The fatigue strength for the Type 308 or 308L material is based on the S-N curve for austenitic stainless steels in Figure I-9.2 of the ASME Code, Section III.

The spider assembly is in the form of a central hub with radial vanes containing cylindrical fingers from which the absorber rods are suspended. Internal groove-like profiles to facilitate

handling tool and drive rod assembly connection are machined into the upper end of the hub. Coil springs inside the spider body absorb the impact energy at the end of a trip insertion. The radial vanes may either be joined to the hub by welding and brazing, and the fingers joined to the vanes by brazing, or the vanes and fingers may be integral with the spider body. A bolt, which holds the springs and retainer, is threaded into the hub within the skirt and welded to prevent loosening while in service.

The components of the spider assembly are made from Types 304, 304L and/or CF-3 (casting equivalent of 304L) stainless steel except for the retainer, which is of Type 630 material, and the springs, which are nickel-chromium-iron Alloy 718.

The absorber rods are fastened securely to the spider. The rods are first threaded into the spider fingers and then secured with a locking device. The end plug below the thread is designed with a reduced section to permit flexing of the rods to correct for small operating or assembly misalignments.

The overall length of the rod cluster control assembly is such that, when the assembly is withdrawn through its full travel, the tips of the absorber rods remain engaged in the guide thimbles so that alignment between rods and thimbles is always maintained. Since the rods are long and slender, they are relatively free to conform to any small misalignments with the guide thimble.

4.2.2.3.2 Gray Rod Cluster Assemblies

Externally the mechanical design of the gray rod cluster assembly is identical to the rod cluster control assembly. In addition, the control rod drive mechanism and the interface with the fuel assemblies and guide thimbles are identical to those of the rod cluster control assembly.

As shown in Figure 4.2-11, the gray rod cluster assemblies consist of 24 rodlets fastened at the top end to a common hub or spider. Geometrically, the gray rod cluster assembly is the same as a rod cluster control assembly except that the absorber material consists of tungsten encapsulated in a nickel-chromium-iron Alloy 718 sleeve and clad with stainless steel cladding which has the same outer diameter as the rod cluster control assembly cladding. The lower portion of the rodlets consists of a stainless steel spacer.

The gray rod cluster assemblies are used in base load operation and load follow maneuvering and provide a mechanical shim to replace the use of changes in the concentration of soluble boron, that is, a chemical shim, normally used for this purpose. The AP1000 uses 53 rod cluster control assemblies and 16 gray rod cluster assemblies.

4.2.2.3.3 Burnable Absorber Assembly

Each burnable absorber assembly consists of discrete burnable absorber rods attached to a hold-down assembly. Figure 4.2-12 shows the burnable absorber assemblies. When needed for nuclear considerations, burnable absorber assemblies are inserted into selected thimbles within fuel assemblies.

The wet annular burnable absorber rods (WABA) consist of pellets of alumina-boron carbide material contained within zirconium alloy tubes. These zirconium alloy tubes, which form the

outer clad for the burnable absorber rod, are plugged, pressurized with helium, and seal-welded at each end to encapsulate the stack of absorber material. The absorber stack length, shown in Figure 4.2-12, is positioned axially within the burnable absorber rod by the use of a zirconium alloy bottom-end spacer as necessary.

The burnable absorber rods in each fuel assembly are grouped and attached together at the top end of the rods to a hold-down assembly by a flat, perforated retaining plate, which fits within the fuel assembly top nozzle and rests on the adapter plate.

The retaining plate and the burnable absorber rods are held down and restrained against vertical motion through a spring pack which is attached to the plate and is compressed by the upper core plate when the reactor upper internals assembly is lowered into the reactor. With this arrangement, the burnable absorber rods cannot be ejected from the core by flow forces. Each rod is attached to the baseplate by a nut that is crimped into place.

4.2.2.3.4 Neutron Source Assemblies

The purpose of a neutron source assembly is to provide a base neutron level to give confidence that the detectors are operational and responding to core multiplication neutrons. For the first core, a neutron source is placed in the reactor to provide a positive neutron count of at least two counts per second on the source range detectors attributable to core neutrons. The detectors, called source range detectors, are used primarily during subcritical modes of core operation.

The source assembly also permits detection of changes in the core multiplication factor during core loading, refueling, and approach to criticality. This can be done since the multiplication factor is related to an inverse function of the detector count rate. Changes in the multiplication factor can be detected during addition of fuel assemblies while loading the core, changes in control rod positions, and changes in boron concentration.

Both primary and secondary neutron source rods are used. The primary source rod, containing a radioactive material, spontaneously emits neutrons during initial core loading, reactor startup, and initial operation of the first core. After the primary source rod decays beyond the desired neutron flux level, neutrons are then supplied by the secondary source rod. The secondary source rod contains a stable material, which is activated during reactor operation. The activation results in the subsequent release of neutrons.

Four source assemblies are typically installed in initial load of the reactor core: two primary source assemblies and two secondary source assemblies. Each primary source assembly contains one primary source rod and a number of burnable absorber rods. Each secondary source assembly contains a symmetrical grouping of secondary source rodlets. Figure 4.2-14 shows the primary source assembly. Figure 4.2-15 shows the secondary source assembly.

Neutron source assemblies are employed at opposite sides of the core. The source assemblies are inserted into the rod cluster control guide thimbles in fuel assemblies at selected locations.

As shown in Figures 4.2-14 and 4.2-15, the source assemblies contain a hold-down assembly identical to that of the burnable absorber assembly.

The primary and secondary source rods both use the same cladding material as the absorber rods. The secondary source rods contain antimony-beryllium pellets stacked to a height of approximately 88 inches. The primary source rods contain capsules of californium (plutonium-beryllium possible alternate) source material and alumina spacers to position the source material within the cladding. The rods in each assembly are fastened at the top end to a hold-down assembly.

The other structural members, except for the springs, are constructed of Type 304, 304L, and 308L stainless steel. The springs exposed to the reactor coolant are nickel-chromium-iron Alloy 718.

4.2.3 Design Evaluation

[*The fuel assemblies, fuel rods, and in-core control components are designed to satisfy the performance and safety criteria of*]* Section 4.2 of the Standard Review Plan, the mechanical design bases of subsection 4.2.1 and [*the Fuel Criteria Evaluation Process per WCAP-12488-A (Reference 1)*]*, and other interfacing nuclear and thermal and hydraulic design bases specified in Sections 4.3 and 4.4.

Effects of Conditions II, III, IV or anticipated transients without trip on fuel integrity are presented in Chapter 15.

The initial step in fuel rod design evaluation for a region of fuel is to determine the limiting rod(s). Limiting rods are defined as those rods whose predicted performance provides the minimum margin to each of the design criteria. For a number of design criteria, the limiting rod is the lead burnup rod of a fuel region. In other instances, it may be the maximum power or the minimum burnup rod. For the most part, no single rod is limiting with respect to all the design criteria.

After identifying the limiting rod(s), an analysis is performed to consider the effects of rod operating history, model uncertainties, and dimensional variations. To verify adherence to the design criteria, the evaluation considers the effects of postulated transient power changes during operation consistent with Conditions I and II. These transient power increases can affect both rod average and local power levels. Parameters considered include rod internal pressure, fuel temperature, clad stress, and clad strain. In fuel rod design analyses, these performance parameters provide the basis for comparison between expected fuel rod behavior and the corresponding design criteria limits.

Fuel rod and assembly models used for the performance evaluations are documented and maintained under an appropriate control system. Material properties used in the design evaluations are given in WCAP-12610 (Reference 5).

4.2.3.1 Cladding

4.2.3.1.1 Vibration and Wear

Fuel rod vibrations are flow induced. The effect of vibration on the fuel assembly and individual fuel rods is minimal. The cyclic stress range associated with deflections of such small magnitude is insignificant and has no effect on the structural integrity of the fuel rod.

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

The reaction force on the grid supports, due to rod vibration motions, is also small and is much less than the spring preload. Adequate fuel clad spring contact is maintained. No significant wear of the clad or grid supports is predicted during the life of the fuel assembly based on out-of-pile flow tests, performance of similarly designed fuel in operating reactors, and design analyses.

Clad fretting and fuel vibration has been experimentally investigated, as shown in WCAP-8278 (Reference 13).

4.2.3.1.2 Fuel Rod Internal Pressure and Cladding Stresses

A burnup-dependent fission gas release model (WCAP-15063-P-A, Revision 1 [Reference 21]) is used in determining the internal gas pressure as a function of irradiation time. The plenum volume of the fuel rod has been designed to provide that the maximum internal pressure of the fuel rod will not exceed the value which would cause:

- The fuel/clad diametral gap to increase during steady-state operation
- Extensive departure from nucleate boiling propagation to occur (Reference 26)

The clad stresses at a constant local fuel rod power are low. Compressive stresses are created by the pressure differential between the coolant pressure and the rod internal gas pressure. Because of the pre-pressurization with helium, the volume average effective stresses are always less than approximately 14,000 psi at the pressurization level used in the AP1000 fuel rod design. Stresses due to the temperature gradient are not included in this average effective stress because thermal stresses are, in general, negative at the clad inside diameter and positive at the clad outside diameter, and their contribution to the clad volume average stress is small. Furthermore, the thermal stress decreases with time during steady-state operation due to stress relaxation. The stress due to pressure differential is highest in the minimum power rod at beginning-of-life due to low internal gas pressure and decreases as rod power increases. Thermal stresses are maximum in the maximum power rod due to the larger temperature gradient and decrease as the rod power is decreased.

The internal gas pressure at beginning-of-life ranges from approximately 200 to 750 psi for typical lead burnup fuel rods. The total tangential stress at the clad inside diameter at beginning-of-life is approximately 19,500 psi compressive (approximately 18,500 psi due to ΔP and approximately 1,000 due to ΔT) for a low-power rod operating at four kilowatts/foot. Total tangential stress is approximately 20,500 psi compressive (approximately 18,000 psi due to ΔP and approximately 2,500 psi due to ΔT) for a high-power rod operating at 10 kilowatts/foot. However, the volume average effective stress at beginning-of-life is between approximately 13,500 psi (high-power rod) and approximately 14,000 psi (low-power rod). These stresses are substantially below even the unirradiated clad yield strength (approximately 55,500 psi) at a typical clad mean operating temperature of 700°F.

Tensile stresses could be created once the clad has come in contact with the pellet. These stresses would be induced by the fuel pellet swelling during irradiation. Swelling of the fuel pellet can result in small clad strains (less than one percent) for expected discharge burnups, but the associated clad stresses are very low because of clad creep (thermal- and irradiation-induced creep). The one percent strain criterion is extremely conservative for fuel-swelling driven clad strain because the strain rate associated with solid fission products swelling is very slow. A detailed discussion of fuel rod performance is given in subsection 4.2.3.3.

4.2.3.1.3 Material and Chemical Evaluation

ZIRLO clad has a high corrosion resistance to the coolant, fuel, and fission products. As shown in WCAP-8183 (Reference 3), there is considerable pressurized water reactor operating experience on the capability of Zircaloy-4 as a clad material. ZIRLO, an advanced zirconium based alloy, has equal or better corrosion resistance than Zircaloy-4 (see WCAP-12610-P-A, [Reference 5]). Controls on fuel fabrication specify maximum moisture levels to preclude clad hydriding.

Metallographic examination of irradiated commercial fuel rods has shown occurrences of fuel/clad chemical interaction. Reaction layers of less than one mil in thickness have been observed between fuel and clad at limited points around the circumference. Metallographic data indicates that this interface layer remains very thin even at high burnup. Thus, there is no indication of propagation of the layer and eventual clad penetration.

Stress corrosion cracking is another postulated phenomenon related to fuel/clad chemical interaction. Out-of-pile tests have shown that in the presence of high clad tensile stresses, large concentrations of iodine can chemically attack the zirconium alloy tubing and may lead to eventual clad cracking. Extensive post-irradiation examination has produced no evidence that this mechanism has been operative in Westinghouse commercial pressurized water reactor fuel.

4.2.3.1.4 Rod Bowing

WCAP-8691 (Reference 14) presents the model used for evaluation of AP1000 fuel rod bowing. This model has been used for bow assessment in 14x14, 15x15, and 17x17 type cores.

4.2.3.1.5 Consequences of Power Coolant Mismatch

Consequences of power coolant mismatch are discussed in Chapter 15.

4.2.3.1.6 Creep Collapse and Creepdown

This subject and the associated irradiation stability of cladding have been evaluated. In WCAP-13589-A (Reference 8), it is shown that current generation Westinghouse fuel is sufficiently stable with respect to fuel densification. Significant axial gaps do not form in the pellet stack, preventing clad collapse from occurring. The design basis of no clad collapse during planned core life is therefore satisfied. Cladding collapse analyses, if required, would be performed using the methods described in WCAP-8377 (Reference 22).

4.2.3.2 Fuel Materials Considerations

Sintered, high-density uranium dioxide fuel reacts only slightly with the clad at core operating temperatures and pressures. In the event of clad defects, the high resistance of uranium dioxide to attack by water protects against fuel deterioration, although limited fuel erosion can occur. The consequences of defects in the clad are greatly reduced by the ability of uranium dioxide to retain fission products, including those which are gaseous or highly volatile.

Observations from several early Westinghouse pressurized water reactors as discussed in WCAP-8218-P-A (Reference 6) have shown that fuel pellets can densify under irradiation to a density higher than the manufactured values. Fuel densification and subsequent settling of the fuel pellets can result in local and distributed gaps in the fuel rods. The densification process is related to the elimination of very small as-fabricated porosity in the fuel during irradiation. Early fuels were intentionally manufactured to low initial density and were undersintered, which resulted in a large fraction of very small pores. Densification behavior in current fuel is controlled by improved manufacturing process controls and by specifying a nominal 95.5 percent initial fuel density, which results in reduced levels of small, densifying porosity.

The evaluation of fuel densification effects and the treatment of fuel swelling and fission gas release are described in WCAP-13589-A (Reference 8) and WCAP-15063-P-A, Revision 1 (Reference 21).

4.2.3.3 Fuel Rod Performance

In the calculation of the steady-state performance of a nuclear fuel rod, the following interacting factors are considered:

- Clad creep and elastic deflection
- Pellet density changes, thermal expansion, gas release, and thermal properties as a function of temperature and fuel burnup
- Internal pressure as a function of fission gas release, rod geometry, and temperature distribution

These effects are evaluated using fuel rod design models, as discussed in WCAP-15063-P-A, Revision 1 (Reference 21), that include appropriate models for time dependent fuel densification. With these interacting factors considered, the model determines the fuel rod performance characteristics for a given rod geometry, power history, and axial power shape. In particular, internal gas pressure, fuel and clad temperatures, and clad deflections are calculated. The fuel rod is divided into several axial sections and radially into a number of annular zones. Fuel density changes are calculated separately for each segment. The effects are integrated to obtain the internal rod pressure.

The initial rod internal pressure is selected to delay fuel/clad mechanical interaction and to avoid the potential for clad flattening. It is limited, however, by the design criteria for the rod internal pressure, as discussed in subsection 4.2.1.3.

The gap conductance between the pellet surface and the clad inner diameter is calculated as a function of the composition, temperature and pressure of the gas mixture, and the gap size or contact pressure between the clad and pellet. After computing the fuel temperature for each pellet zone, the fractional fission gas release is assessed using an empirical model derived from experimental data, as detailed in WCAP-15063-P-A, Revision 1 (Reference 21). The total amount of gas released is based on the average fractional release within each axial and radial zone and the gas generation rate, which, in turn, is a function of burnup. Finally, the gas released is summed over the zones, and the pressure is calculated.

The model shows close agreement in fit for a variety of published and proprietary data on fission gas release, fuel temperatures, and clad deflections, as detailed in WCAP-15063-P-A, Revision 1 (Reference 21). These data include variations in power, time, fuel density, and geometry.

4.2.3.3.1 Fuel/Cladding Mechanical Interaction

One factor in fuel element duty is potential mechanical interaction of the fuel and clad. This fuel/clad interaction produces cyclic stresses and strains in the clad, and these, in turn, reduce clad life. The reduction of fuel/clad interaction is therefore a goal of design. The technology for using pre-pressurized fuel rods in Westinghouse pressurized water reactors has been developed to further this objective.

The gap between the fuel and clad is initially sufficient to prevent hard contact between the two. However, during power operation a gradual compressive creep of the clad onto the fuel pellet occurs due to the external pressure exerted on the rod by the coolant. Clad compressive creep eventually results in fuel/clad contact. Once fuel/clad contact occurs, changes in power level result in changes in clad stresses and strains. By using pre-pressurized fuel rods to partially offset the effect of the coolant external pressure, the rate of clad creep toward the surface of the fuel is reduced. Fuel rod pre-pressurization delays the time at which fuel/clad contact occurs and, hence, significantly reduces the extent of cyclic stresses and strains experienced by the clad both before and after fuel/clad contact. These factors result in an increase in the fatigue life margin of the clad and lead to greater clad reliability.

A two-dimensional (r,θ) finite element model has been established to investigate the effects of radial pellet cracks on stress concentrations in the clad. Stress concentration herein is defined as the difference between the maximum clad stress in the θ direction and the mean clad stress. The first case has the fuel and clad in mechanical equilibrium; and, as a result, the stress in the clad is close to zero. In subsequent cases the pellet power is increased in steps and the resultant fuel thermal expansion imposes tensile stress in the clad.

In addition to uniform clad stresses, stress concentrations develop in the clad adjacent to radial cracks in the pellet. These radial cracks have a tendency to open during a power increase, but the frictional forces between fuel and clad oppose the opening of these cracks and result in localized increases in clad stress. As the power is further increased, large tensile stresses exceed the ultimate tensile strength of uranium dioxide and additional cracks in the fuel pellet are created, limiting the magnitude of the stress concentration in the clad.

As part of the standard fuel rod design analysis, the maximum stress concentration evaluated from finite element calculations is added to the volume-averaged effective stress in the clad as determined from one-dimensional stress/strain calculations. The resultant clad stress is then compared to the temperature-dependent cladding yield stress to confirm that the stress/strain criteria are satisfied.

The transient evaluation method is described in the following paragraphs.

Pellet thermal expansion due to power increases is considered the only mechanism by which significant stresses and strains can be imposed on the clad.

Power increases in commercial reactors can result from fuel shuffling (for example, a fuel assembly positioned near the core center for cycle 2 operation after operating near the periphery during cycle 1), reactor power escalation following extended reduced power operation, and full-length control rod movement. In the mechanical design model, lead rods are depleted using best-estimate power histories as determined by core physics calculations. During burnup, the amount of diametral gap closure is evaluated based upon the pellet expansion cracking model, clad creep model, and fuel swelling model. At various times during the depletion, the power is increased locally in the rod to the burnup-dependent attainable power density as determined by core physics calculations. The radial, tangential, and axial clad stresses resulting from the power increase are combined into a volume average effective clad stress.

The von Mises criterion is used to determine whether the clad yield stress has been exceeded. This criterion states that an isotropic material in multi-axial stress will begin to yield plastically when the effective stress exceeds the yield stress as determined by an axial tensile test. The yield stress correlation is that for irradiated cladding, since fuel/clad interaction occurs at high burnup. In applying this criterion, the effective stress is increased by an allowance which accounts for stress concentrations in the clad adjacent to radial cracks in the pellet, prior to the comparison with the yield stress. This allowance was evaluated using a two-dimensional (r,θ) finite element model.

Slow transient power increases can result in large clad strains without exceeding the clad yield stress because of clad creep and stress relaxation. Therefore, in addition to the yield stress criterion, a criterion on allowable clad strain is necessary. Based upon high strain rate burst and tensile test data on irradiated tubing, one percent strain was determined to be a conservative lower limit on irradiated clad ductility and that was adopted as a design criterion.

In addition to the mechanical design models and design criteria, the AP1000 fuel rod design relies on performance data accumulated through transient power test programs in experimental and commercial reactors, and through normal operation in commercial reactors.

It is recognized that a possible limitation to the satisfactory behavior of the fuel rods in a reactor subjected to daily load follow is the failure of the cladding by low-cycle strain fatigue. During their normal residence time in the reactor, the fuel rods may be subjected to on the order of 1000 load follow cycles, with typical changes in power level from 50 to 100 percent of their steady-state values.

The assessment of the fatigue life of the fuel rod cladding is subjected to considerable uncertainty because of the difficulty of evaluating the strain range which results from the cyclic interaction of the fuel pellets and cladding. This difficulty arises, for example, from such highly unpredictable phenomena as pellet cracking, fragmentation, and relocation. Westinghouse investigated this particular phenomenon both analytically and experimentally. Strain fatigue tests on irradiated and nonirradiated hydrided Zircaloy-4 cladding were performed. These tests permitted the definition of a conservative fatigue-life limit and recommendation of a methodology to treat the strain fatigue evaluation of the Westinghouse-referenced fuel rod designs. (See WCAP-9500-P-A, Reference 15.)

Successful load follow operation has been performed on several reactors. There was no significant coolant activity increase that could be associated with the load follow mode of operation.

The Westinghouse analytical approach to strain fatigue is based on a comprehensive review of the available strain fatigue models. The review included the Langer-O'Donnell model (Reference 16) the Yao-Munse model, and the Manson-Halford model. Upon completion of this review, and using the results of the Westinghouse experimental programs as documented in WCAP-9500-P-A (Reference 15), it was concluded that the approach defined by Langer-O'Donnell would be retained and the empirical factors of their correlation modified to conservatively bound the results of the Westinghouse testing program.

The design equations followed the concept for the fatigue design criterion according to the ASME Code, Section III:

- The calculated pseudo stress amplitude (S_a) has to be multiplied by a factor of two to obtain the allowable number of cycles (N_f).
- The allowable cycles for a given S_a is five percent of N_f or a safety factor of 20 on cycles.

The lesser of the two allowable numbers of cycles is selected. The cumulative fatigue life fraction is then computed as:

$$\sum_{1}^{k} \frac{n_k}{N_{fk}} \leq 1$$

where:

n_k = number of diurnal cycles of mode k.

N_{fk} = number of allowable cycles.

4.2.3.3.2 Irradiation Experience

Westinghouse fuel operational experience is presented in CENPD-404-P-A (Reference 27). Additional test assembly and test rod experience is given in WCAP-10125-P-A (Reference 2).

4.2.3.3.3 Fuel and Cladding Temperature

The methods used for evaluation of fuel rod temperatures are presented in subsection 4.4.2.11.

4.2.3.3.4 Potentially Damaging Temperature Effects During Transients

The fuel rod experiences many operational transients (intentional maneuvers) during its residence in the core. A number of thermal effects must be considered when analyzing the fuel rod performance.

The clad can be in contact with the fuel pellet at some time in the fuel lifetime. Clad/pellet interaction occurs if the fuel pellet temperature is increased after the clad is in contact with the pellet. Clad/pellet interaction is discussed in subsection 4.2.3.3.1.

Clad flattening has been observed in some operating power reactors. This is no longer a concern because clad flattening is precluded during the fuel residence in the core (subsection 4.2.3.1) by the use of stable fuel.

Potential differential thermal expansion between the fuel rods and the guide thimbles during a transient is considered in the design. Excessive bowing of the fuel rods is precluded because the grid assemblies allow axial movement of the fuel rods relative to the grids. Specifically, thermal expansion of the fuel rods is considered in the grid design so that axial loads imposed on the fuel rods during a thermal transient will not result in excessively bowed fuel rods.

4.2.3.3.5 Fuel Element Burnout and Potential Energy Release

As discussed in subsection 4.4.2.2, the core is protected from departure from nucleate boiling over the full range of possible operating conditions. In the extremely unlikely event that departure from nucleate boiling should occur, the clad temperature will rise due to the steam blanketing at the rod surface and the consequent degradation in heat transfer. During this time there is a potential for chemical reaction between the cladding and the coolant. However, because of the relatively good film boiling heat transfer following departure from nucleate boiling, the energy release resulting from this reaction is insignificant compared to the power produced by the fuel.

4.2.3.3.6 Coolant Flow Blockage Effects on Fuel Rods

The coolant flow blockage effects on fuel rods are presented in subsection 4.4.4.7.

4.2.3.4 Spacer Grids

The coolant flow channels are established and maintained by the structure composed of grids and guide thimbles. The lateral spacing between fuel rods is provided and controlled by the support dimples of adjacent grid cells. Contact of the fuel rods on the dimples is maintained through the clamping force of the grid springs. Lateral motion of the fuel rods is opposed by the spring force and the internal moments generated between the spring and the support dimples. Grid testing is discussed in WCAP-8236 (Reference 17) and WCAP-10444-P-A (Reference 11).

4.2.3.5 Fuel Assembly

4.2.3.5.1 Stresses and Deflections

The fuel assembly component stress levels are limited by the design. For example, stresses in the fuel rod due to thermal expansion and zirconium alloy irradiation growth are limited by the relative motion of the rod as it slips over the grid spring and dimple surfaces. Clearances between the fuel rod ends and nozzles are provided so that zirconium alloy irradiation growth does not result in rod end interference. Stresses in the fuel assembly caused by tripping of the rod cluster control assembly have little influence on fatigue usage margin because of the small number of events during the life of an assembly. Assembly components and prototype

fuel assemblies made from production parts have been subjected to structural tests to verify that the design bases requirements are met.

The fuel assembly design loads for shipping have been established at 4 g axial and 6 g lateral. Accelerometers are permanently placed in the shipping cask to monitor and detect fuel assembly accelerations that would exceed the criteria. Experience indicates that loads that exceed the allowable limits rarely occur. Exceeding the limits requires reinspection of the fuel assembly for damage. Tests on various fuel assembly components, such as the grid assembly, sleeves, inserts, and structure joints, have been performed to confirm that the shipping design limits do not result in impairment of fuel assembly function. Seismic analysis methodology of the fuel assembly is presented in WCAP-8236 (Reference 17), WCAP-9401-P-A (Reference 18), and WCAP-10444-P-A (Reference 11).

To demonstrate that the fuel assemblies will maintain a geometry that is capable of being cooled under the worst-case accident Condition IV event, a plant specific or bounding seismic analysis is performed.

The fuel assembly response resulting from safe shutdown earthquake condition is analyzed using time-history numerical techniques. The vessel motion for this type of event primarily causes lateral loads on the reactor core. Consequently, the methodology and analytical procedures as described in WCAP-8236 (Reference 17) and WCAP-9401-P-A (Reference 18) are used to assess the fuel assembly deflections and impact forces.

The motions of the reactor internals upper and lower core plates and the core barrel at the upper core plate elevation, which are simultaneously applied to simulate the reactor core input motion, are obtained from the time-history analysis of the reactor vessel and internals. The fuel assembly response, namely the displacements and impact forces, is obtained with the reactor core model. Similar dynamic analyses of the core were performed using reactor internals motions indicative of the postulated pipe rupture. Scenarios regarding breaches in the pressure boundary are investigated to determine the most limiting structural loads for the fuel assembly. The application of leak-before-break limits the size of the pipe rupture loads for which the fuel assemblies must be analyzed. The pipe rupture used in the fuel assembly analysis is the largest pipe connected to the reactor coolant system which does not satisfy the leak-before-break criteria. Subsection 3.6.3 discusses mechanistic pipe break.

4.2.3.5.1.1 Grid Analyses

The maximum grid impact force obtained from seismic analyses is less than the allowable grid strength. With respect to the guidelines of Appendix A of the Standard Review Plan, Section 4.2, Westinghouse has demonstrated that a simultaneous safe shutdown earthquake and pipe rupture event is highly unlikely. The fatigue cycles, crack initiation, and crack growth due to normal operating and seismic events will not realistically lead to a pipe rupture. More information is available in WCAP-9283 (Reference 19).

Based on the deterministic fracture mechanics evaluation of small flaws in piping components, Westinghouse has demonstrated that the dynamic affects of a large pipe rupture in the primary coolant piping system for the AP1000 design does not have to be considered.

A design basis for the piping design in the AP1000 is that the reactor coolant loop and surge lines will satisfy the leak-before-break criteria for mechanistic pipe break. In addition, the

piping connected to the reactor coolant system that is six inch nominal diameter or larger is evaluated for leak-before-break. The result of a pipe leakage event consistent with the mechanistic pipe break evaluation would be to impose insignificant asymmetric loadings on the reactor core system. Thus, fuel assembly grid loads due to large pipe ruptures are unrealistic and, as such, are not included in the analysis.

The pressure boundary integrity for numerous branch lines is analyzed to determine the most limiting break of a line not qualified for leak-before-break for the dynamic loading of the reactor core. Grid loads resulting from a combined seismic and pipe rupture event do not cause unacceptable grid deformation as to preclude a core coolable geometry.

4.2.3.5.1.2 Nongrid Analyses

The stresses induced in the various fuel assembly nongrid components are assessed based on the most limiting seismic condition. The fuel assembly axial forces resulting from the hold-down spring load together with its own weight distribution are the primary sources of the stresses in the guide thimbles and fuel assembly nozzles. The fuel rod accident induced stresses, which are generally very small, are caused by bending due to the fuel assembly deflections during a seismic event. The seismic-induced stresses are compared with the allowable stress limits for the fuel assembly major components. The component stresses, which include normal operating stresses, are below the established allowable limits. Consequently, the structural designs of the fuel assembly components are acceptable for the design basis accident conditions for the AP1000.

4.2.3.5.2 Dimensional Stability

Localized yielding and slight deformation in some fuel assembly components are allowed to occur during a Condition III or IV event. The maximum permanent deflection, or deformations, do not result in any violation of the functional requirements of the fuel assembly.

4.2.3.6 Reactivity Control Assemblies and Burnable Absorber Rods

4.2.3.6.1 Internal Pressure and Cladding Stresses during Normal, Transient, and Accident Conditions

The designs of the burnable absorber, source, and gray rods provide a sufficient cold void volume to accommodate the internal pressure increase during operation. This is not a concern for the rod cluster control assembly absorber rodlets because no significant amount of gas is released by the silver-indium-cadmium absorber material.

For the discrete burnable absorber rod, there is sufficient cold void volume to limit the internal pressure to a value, which satisfies the design criteria. For the source rods and gray rods, a void volume is provided within the rod to limit the maximum internal pressure increase at end-of-life. Figures 4.2-14 and 4.2-15 detail the primary and secondary source assemblies and Figure 4.2-11 details the gray rod cluster assembly.

During normal transient and accident conditions, the void volume limits the internal pressures to values that satisfy the criteria in subsection 4.2.1.6. These limits are established not only to prevent the peak stresses from reaching unacceptable values, but also to limit the amplitude

of the oscillatory stress component in consideration of the fatigue characteristics of the materials.

Rod, guide thimble, and dashpot flow analyses indicate that the flow is sufficient to prevent coolant boiling within the guide thimble. Therefore, clad temperatures at which the clad material has adequate strength to resist coolant operating pressures and rod internal pressures are maintained.

4.2.3.6.2 Thermal Stability of the Absorber Material, Including Changes and Thermal Expansion

The radial and axial temperature profiles within the source and absorber rods are determined by considering gap conductance, thermal expansion, neutron or gamma heating of the contained material as well as gamma heating of the clad.

The maximum temperatures of the silver-indium-cadmium RCCA or tungsten GRCA absorber materials are calculated and found to be significantly less than the material melting point and found to occur axially at only the highest flux region. The mechanical and thermal expansion properties of the silver-indium-cadmium absorber material are discussed in WCAP-9179 (Reference 4). The mechanical and thermal expansion properties of the tungsten absorber material are discussed in WCAP-16943-P-A (Reference 25).

The wet annular burnable absorber (WABA) assemblies are used in the first core. The maximum temperature of the alumina-boron carbide burnable absorber pellet is expected to be less than 1200°F which takes place following the initial power ascent. As the operating cycle proceeds, the burnable absorber pellet temperature decreases due to a reduction in heat generation due to boron depletion and better gap conduction as the helium produced diffuses into the gap.

Sufficient diametral and end clearances have been provided in the neutron absorber, burnable absorber, and source rods to accommodate the relative thermal expansions between the enclosed material and the surrounding clad and end plug.

4.2.3.6.3 Irradiation Stability of the Absorber Material, Taking into Consideration Gas Release and Swelling

The irradiation stability of the silver-indium-cadmium absorber material is discussed in WCAP-9179 (Reference 4). Irradiation produces no deleterious effects in the absorber material. The irradiation stability of the tungsten absorber material is discussed in WCAP-16943-P-A (Reference 25).

As mentioned in subsection 4.2.3.6.1, gas release is not a concern for the rod cluster control rod material because no gas is produced by the absorber material. Sufficient diametral and end clearances are provided to accommodate any potential expansion and/or swelling of the absorber material for both RCCA and GRCA absorber rods.

Irradiation produces no deleterious effects in the tungsten absorber material of the gray rodlets. Some minor cracking of the tungsten material may occur, but this cracking does not affect the absorber column geometric stability due to the small clearance between absorber and sleeve (Reference 25).

The alumina-boron carbide burnable absorber pellets are designed such that gross swelling or crumbling of the pellets is not predicted to occur during reactor operation. Some minor cracking of the pellets may occur, but this cracking should not affect the overall absorber and stack integrity.

4.2.3.6.4 Potential for Chemical Interaction, Including Possible Waterlogging Rupture

The structural materials selected have good resistance to irradiation damage and are compatible with the reactor environment.

Corrosion of the materials exposed to the coolant is quite low, and proper control of chloride and oxygen in the coolant minimizes potential for the occurrence of stress corrosion. The potential for the interference with rod cluster control assembly movement due to possible corrosion phenomena is very low.

Waterlogging rupture is not a failure mechanism associated with the AP1000 control rods. Furthermore, a breach of the cladding for any postulated reason does not result in serious consequences.

The silver-indium-cadmium absorber material is relatively inert and will remain inert even when subjected to high coolant velocity regions. Rapid loss of reactivity control material will not occur. Test results detailed in WCAP-9179 (Reference 4) concluded that additions of indium and cadmium to silver, in the amounts to form the silver-indium-cadmium absorber material composition, result in small corrosion rates.

In the unlikely event of GRCA rod cladding breach, loss of absorber material will not occur because the inner sleeve encapsulates the tungsten absorber (WCAP-16943-P-A, Reference 25).

For the discrete burnable absorber, in the unlikely event that the zirconium alloy clad is breached, the boron carbide in the affected rod(s) could be leached out by the coolant water. If this occurred early, in-core instruments could detect large peaking factor changes, and corrective action would be taken, if warranted. A postulated clad breach after substantial irradiation would have no significant effect on peaking factors since the boron will have been depleted. Breaching of the zirconium alloy clad by internal hydriding is not expected due to moisture controls employed during fabrication. Rods of this design have performed very well with no failures observed.

4.2.4 Testing and Inspection Plan

4.2.4.1 Quality Assurance Program

The Quality Assurance Program Plan of the Westinghouse Commercial Nuclear Fuel Division for the AP1000 is summarized in Chapter 17.

The program provides for control over activities affecting product quality, commencing with design and development and continuing through procurement, materials handling, fabrication, testing and inspection, storage, and transportation. The program also provides for the indoctrination and training of personnel and for the auditing of activities affecting product quality through a formal auditing program.

Westinghouse drawings and product, process, and material specifications identify the inspections to be performed.

4.2.4.2 Quality Control

Quality control philosophy is generally based on the following inspections being performed to a 95 percent confidence that at least 95 percent of the product meets specification, unless otherwise noted.

4.2.4.2.1 Fuel System Components and Parts

The characteristics inspected depend on the component parts. The quality control program includes dimensional and visual examinations, check audits of test reports, material certification, and nondestructive examination, such as X-ray and ultrasonic.

The material used in the AP1000 core is accepted and released by Quality Control.

4.2.4.2.2 Pellets

Inspection is performed for dimensional characteristics such as diameter, density, length, and squareness of ends. Additional visual inspections are performed for cracks, chips, and surface conditions according to approved standards.

Density is determined in terms of weight per unit length and is plotted on zone charts used in controlling the process. Chemical analyses are taken on a specified sample basis throughout pellet production.

4.2.4.2.3 Rod Inspection

Fuel rod, rod cluster control rod, discrete burnable absorber rod, and source rod inspections consists of the following nondestructive examination techniques and methods, as applicable:

- Each rod is leak tested using a calibrated mass spectrometer, with helium being the detectable gas.
- Rod welds are inspected by ultrasonic test or X-ray in accordance with a qualified technique and Westinghouse specifications meeting the requirements of ASTM-E-142-86 (Reference 20).
- Rods are dimensionally inspected prior to final release. The requirements include such items as length, camber, and visual appearance.
- Fuel rods are inspected by gamma scanning or other approved methods, as discussed in subsection 4.2.4.5, to confirm proper plenum dimensions.
- Fuel rods are inspected by gamma scanning, or other approved methods, as discussed in subsection 4.2.4.5, to confirm that no significant gaps exist between pellets.

- Fuel rods are actively and/or passively gamma scanned to verify enrichment control prior to acceptance for assembly loading.
- Traceability of rods and associated rod components is established by quality control.

4.2.4.2.4 Assemblies

Each fuel rod, control rod, burnable absorber rod and source rod assembly is inspected for compliance with drawing and/or specification requirements. Other in-core control component inspection and specification requirements are given in subsection 4.2.4.4.

4.2.4.2.5 Other Inspections

The following inspections are performed as part of the routine inspection operation:

- Tool and gauge inspection and control, including standardization to primary and/or secondary working standards. Tool inspection is performed at prescribed intervals on serialized tools. Complete records are kept of calibration and conditions of tools.
- Audits are performed of inspection activities and records to confirm that prescribed methods are followed and that records are correct and properly maintained.
- Surveillance inspection, where appropriate, and audits of outside contractors are performed to confirm conformance with specified requirements.

4.2.4.2.6 Process Control

To prevent the possibility of mixing enrichments during fuel manufacture and assembly, strict enrichment segregation and other process controls are exercised.

The uranium dioxide powder is kept in sealed containers. The contents are fully identified both by descriptive tagging and unique barcode numbers. A quality control identification tag completely describing the contents is affixed to the containers before transfer to powder storage. Isotopic content is confirmed by analysis.

Powder withdrawal from storage can be made by only one authorized group, which directs the powder to the correct pellet production line. The pellet production lines are physically separated from each other, and pellets of only a single nominal enrichment and density are produced in a given production line at any given time.

Finished pellets are placed on trays identified with the same color code as the powder containers and transferred to segregated storage racks within the confines of the pelleting area. Samples from each pellet lot are tested for isotopic content and impurity levels prior to acceptance by quality control. Physical barriers are used to prevent mixing of pellets of different nominal densities and enrichments in the pellet storage area. Unused powder and substandard pellets are returned to storage in the original color-coded containers.

Loading of pellets into the clad is performed in isolated production lines; only one density and enrichment (with possible exception for top and bottom (axial blanket) zones) are loaded on a line at a time.

A serialized traceability code is placed on each fuel tube, which identifies the contract and enrichment. The end plugs are inserted and then welded (in an inert gas atmosphere) to seal the tube. The fuel tube remains coded and traceability identified until just prior to installation in the fuel assembly.

Similar traceability is provided for wet annular burnable absorber, source, and control rods, as required.

4.2.4.3 Letdown Radiation Monitoring

Radiation monitoring of the reactor coolant is made by grab samples and laboratory analysis of the primary coolant. Refer to information presented in subsections 9.3.3 and 9.3.6, and Table 9.3.3-1.

4.2.4.4 In-core Control Component Testing and Inspection

Tests and inspections are performed on each reactivity control component to verify the mechanical characteristics. In the case of the rod cluster control assembly, prototype testing has been conducted. Manufacturing test/inspections and functional testing at the plant site are both performed.

During the component manufacturing phase, the following requirements apply to the reactivity control components to provide the proper functioning during reactor operation:

- Materials are procured to specifications to attain the desired standard of quality.
- Spider assemblies with brazed and welded vanes and fingers are proof-tested by applying a 5000-pound load to the spider body, so that approximately 310 pounds is applied to each vane. This proof load provides a bending moment at the spider body approximately equivalent to 1.4 times the load caused by the acceleration imposed by the control rod drive mechanism.
- Rods are checked for integrity by the applicable nondestructive methods described in subsection 4.2.4.2.3.
- To confirm proper fit with the fuel assembly, the rod cluster control, discrete burnable absorber, and source assemblies are installed in the fuel assembly and checked for binding in the dry condition.

The rod cluster control assemblies and gray rod cluster assemblies are also functionally tested, following core loading but prior to criticality, to demonstrate reliable operation of the assemblies. Each assembly is operated (and tripped) one time at full-flow/hot conditions. In addition, any assembly that has a drop time greater than a two sigma limit from the average rod drop time is subjected to additional rod drops to confirm drop time. Thus, each assembly is sufficiently tested to confirm proper functioning and operation.

To demonstrate continuous free movement of the rod cluster control assemblies, and gray rod cluster assemblies, and to provide acceptable core power distributions during operations, partial movement checks are performed as required by the technical specifications. In

addition, periodic drop tests of the rod cluster control assemblies are performed at each refueling shutdown to demonstrate continued ability to meet trip time requirements.

If a rod cluster control assembly and/or gray rod cluster assembly cannot be moved by its mechanism, and is determined to be untrippable, adjustments in the boron concentration of the coolant provide that adequate shutdown margin will be achieved following a trip. Thus, inability to move one assembly can be tolerated until the reactor can be safely taken to Mode 3.

4.2.4.5 Tests and Inspections by Others

For tests and inspections performed by others, Westinghouse reviews and approves the quality control procedures, and inspection plans to be utilized to confirm that they are equivalent to the description provided in subsections 4.2.4.1 through 4.2.4.4 and are performed properly to meet Westinghouse requirements.

4.2.4.6 Inservice Surveillance

As detailed in CENPD-404-P-A (Reference 27), significant 17x17 fuel assembly operating experience has been obtained. A surveillance program is expected to be established for the AP1000 for inspection of post-irradiated fuel assemblies. This surveillance program will establish the schedule, guidelines, and inspection criteria for conducting visual inspection of post-irradiated fuel assemblies and/or insert components. The surveillance program includes a visual examination of some discharged fuel assemblies from each refueling. This program also includes criteria for additional inspection requirements for post-irradiated fuel assemblies if unusual characteristics are noticed in the visual inspection or if plant instrumentation and subsequent laboratory analysis indicates gross failed fuel. The post-irradiated fuel surveillance program will address disposition of fuel assemblies and/or insert components receiving an unsatisfactory visual inspection. Those post-irradiated fuel assemblies receiving an unsatisfactory visual inspection are not reinserted into the core until a more detailed inspection and/or evaluation can be performed.

4.2.4.7 Onsite Inspection

Written procedures are used for the post-shipment inspection of the new fuel assemblies in addition to reactivity control and source components. Fuel handling procedures specify the sequence in which handling and inspection take place.

Loaded fuel containers, when received onsite, are externally inspected to confirm that labels and markings are intact and security seals are unbroken. After the containers are opened, the shock indicators attached to the suspended internals are inspected to determine whether movement during transit exceeded design limitations.

Following removal of the fuel assembly from the container in accordance with detailed procedures, the fuel assembly plastic wrapper is examined for evidence of damage. The polyethylene wrapper is then removed, and a visual inspection of the entire fuel assembly is performed.

Control rod, gray rod, secondary source rod and discrete burnable absorber rod assemblies are usually shipped in fuel assemblies. They are inspected either prior to removal of the fuel assembly from the container or after the fuel assemblies are placed in the new fuel storage racks. The control rod assembly is withdrawn a few inches from the fuel assembly to confirm free and unrestricted movement, and the exposed section is visually inspected for mechanical integrity, replaced in the fuel assembly, and stored with the fuel assembly. Control rod, secondary source or discrete burnable absorber assemblies may be stored separately or within fuel assemblies in the new fuel storage area.

4.2.5 Combined License Information

The Combined License information requested in this subsection has been completely addressed in APP-GW-GLR-059 (Reference 24), and the applicable changes have been incorporated into the DCD. No additional work is required by the Combined License applicant to address the Combined License information requested in this subsection.

The following words represent the original Combined License Information Item commitment, which has been addressed as discussed above:

> Combined License applicants referencing the AP1000 certified design will address changes to the reference design of the fuel, burnable absorber rods, rod cluster control assemblies, or initial core design from that presented in the DCD.

4.2.6 References

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*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

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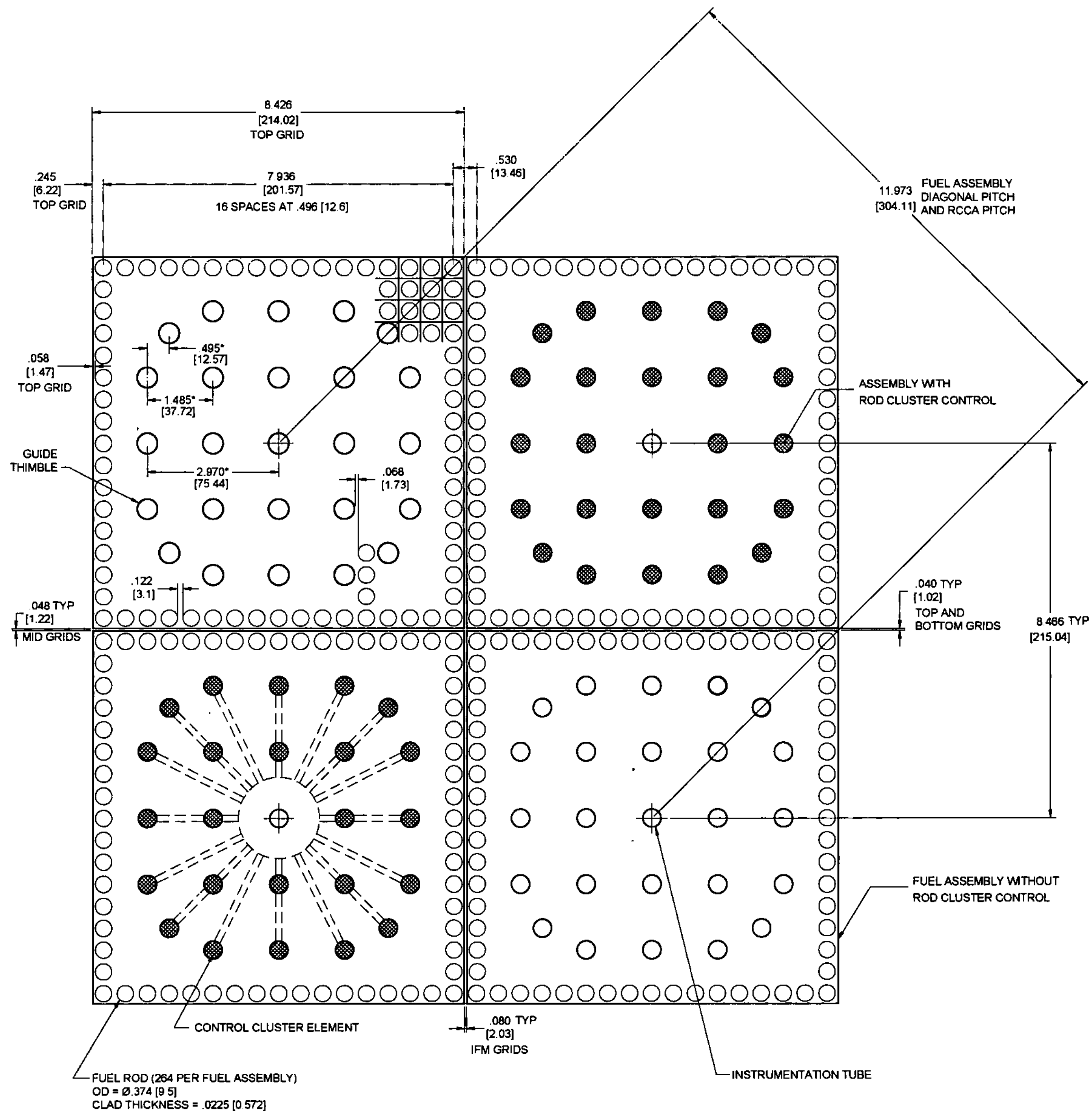


PRIMARY DIMENSIONS ARE IN INCHES (NOMINAL)
SECONDARY DIMENSIONS ARE IN MILLIMETERS

* GUIDE THIMBLE LOCATIONS
AT TOP NOZZLE ADAPTER PLATE

Figure 4.2-1

Fuel Assembly Cross-Section

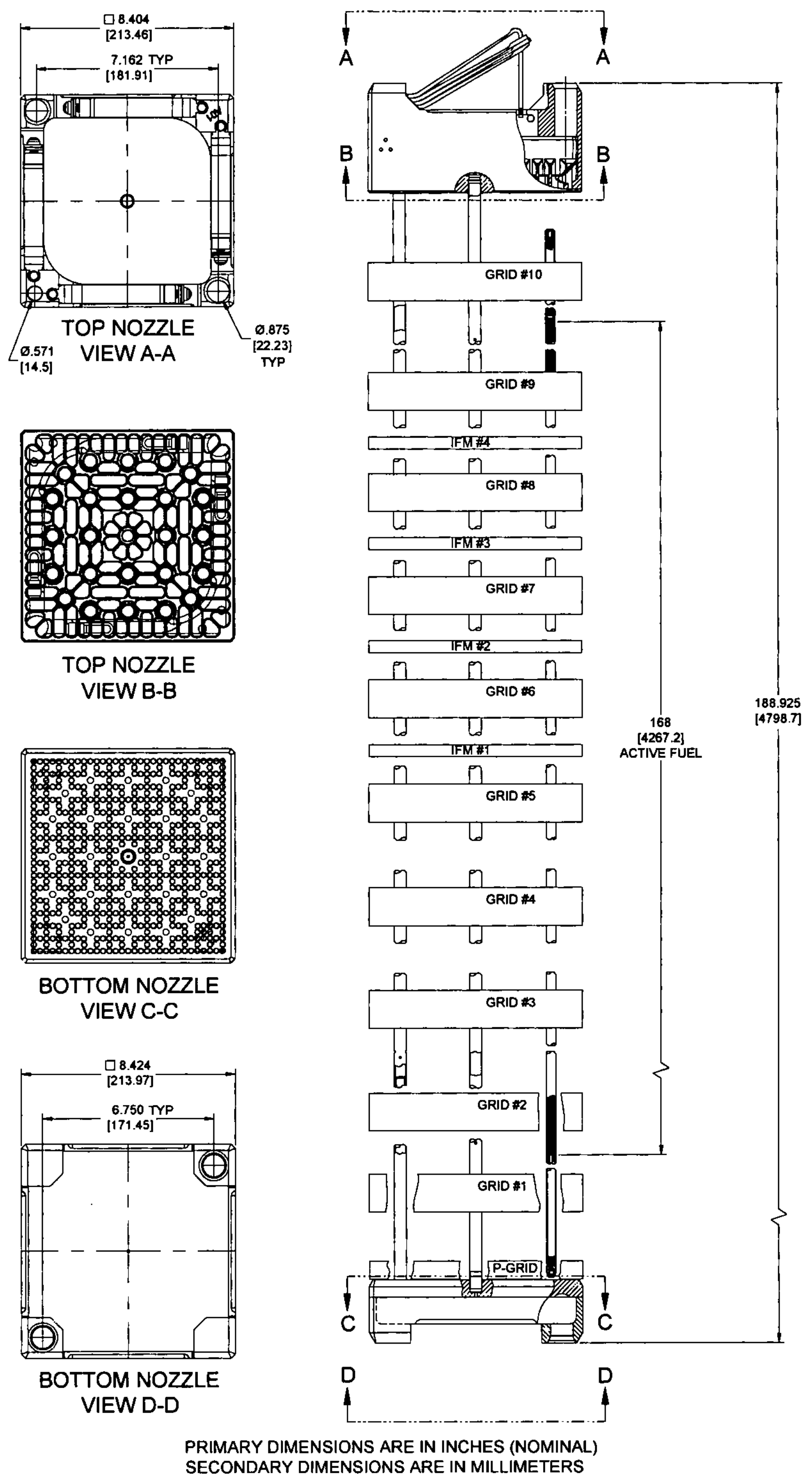


Figure 4.2-2

Fuel Assembly Outline

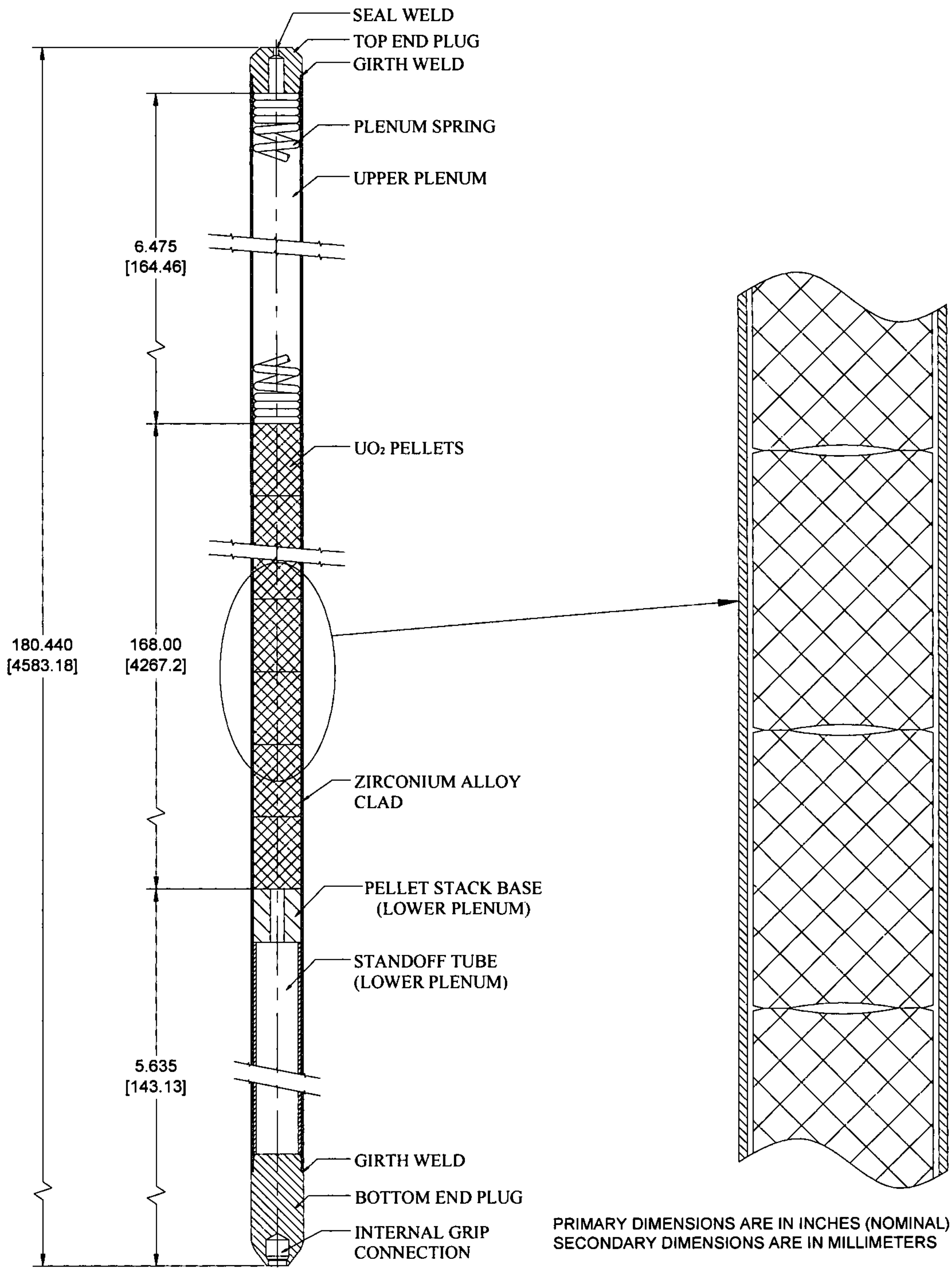


Figure 4.2-3

Fuel Rod Schematic

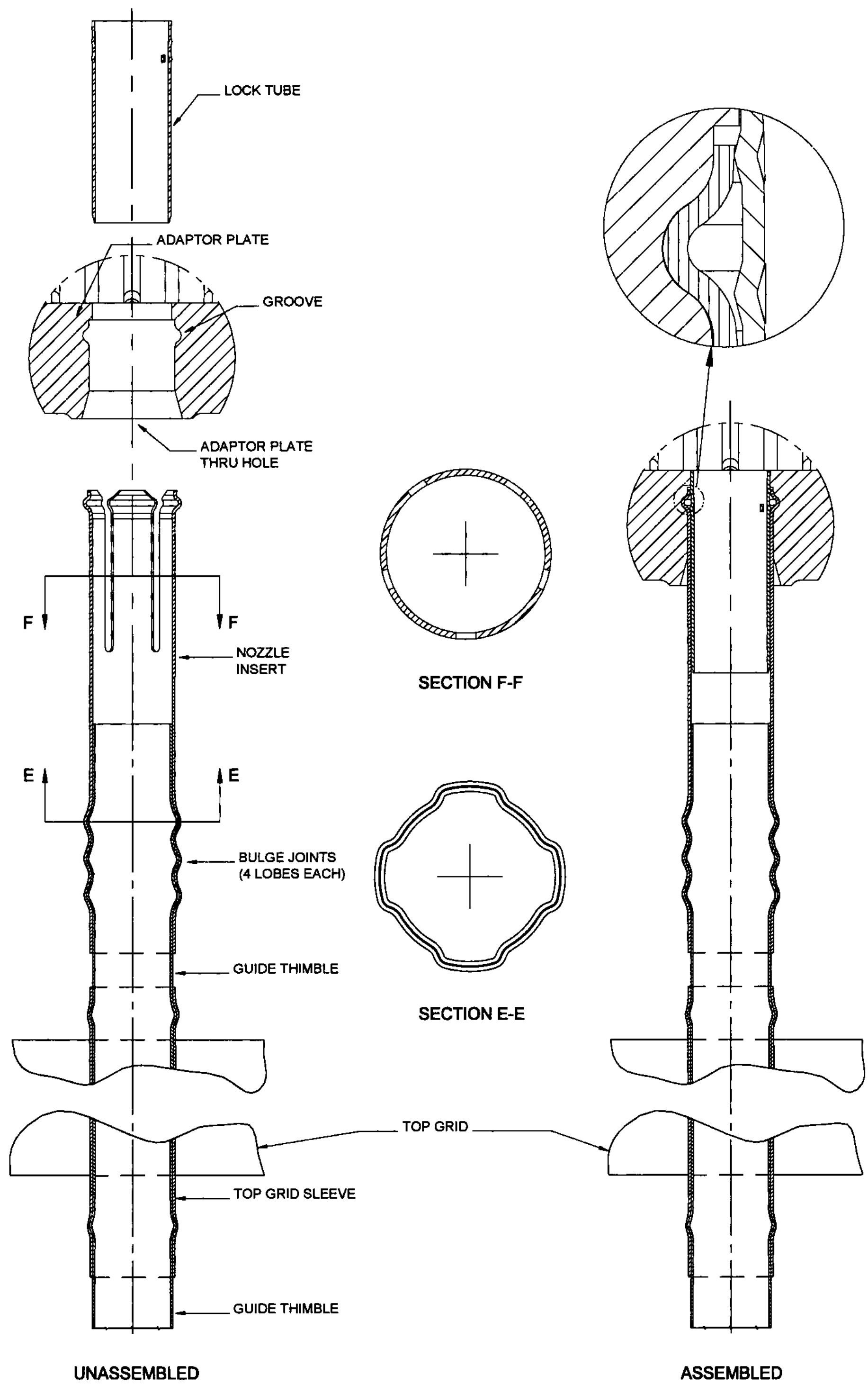


Figure 4.2-4

Top Grid Sleeve Detail

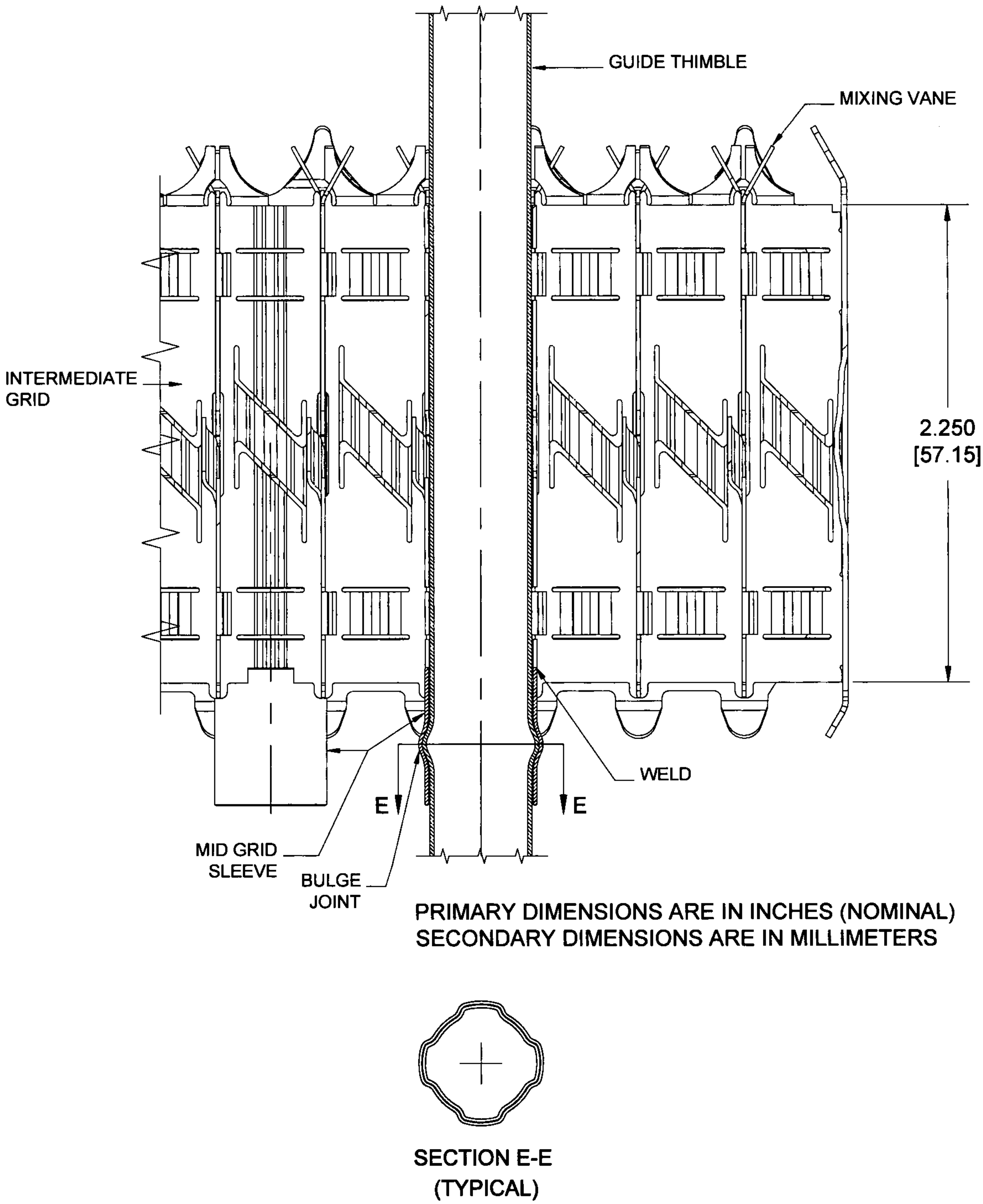


Figure 4.2-5

Intermediate Grid to Thimble Attachment Joint

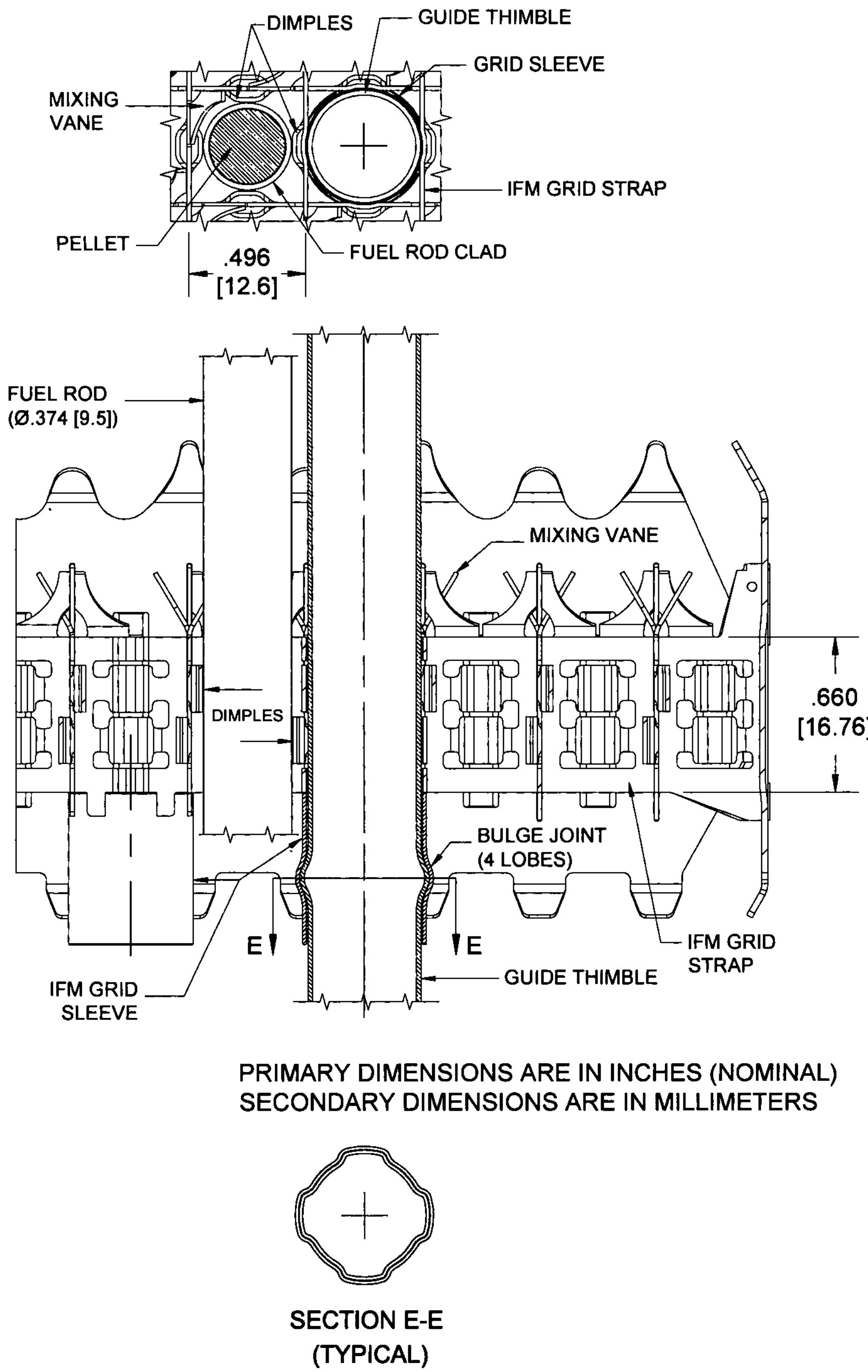


Figure 4.2-6

Intermediate Flow Mixer Grid to Thimble Attachment

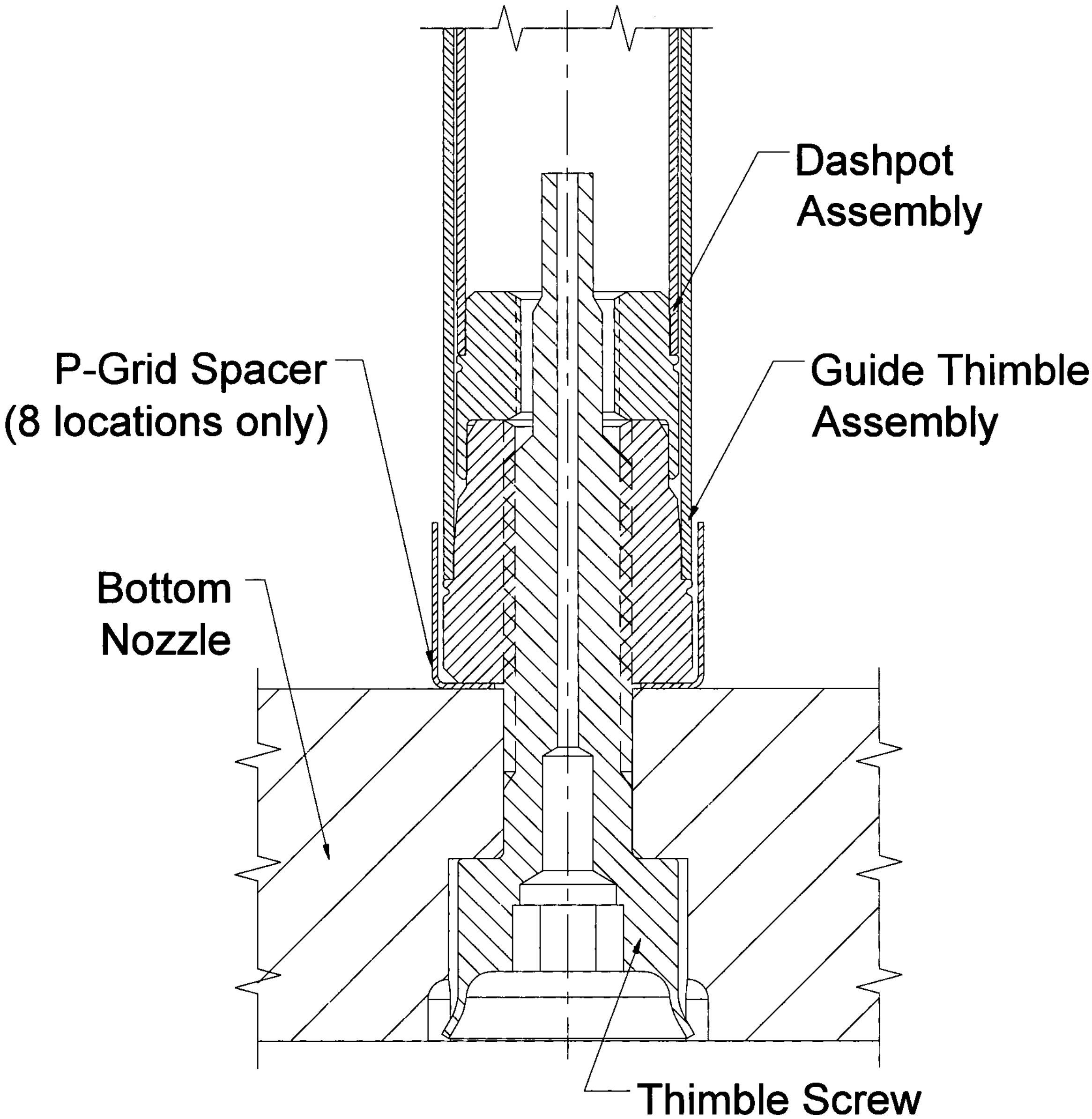


Figure 4.2-7

Grid Thimble to Bottom Nozzle Joint

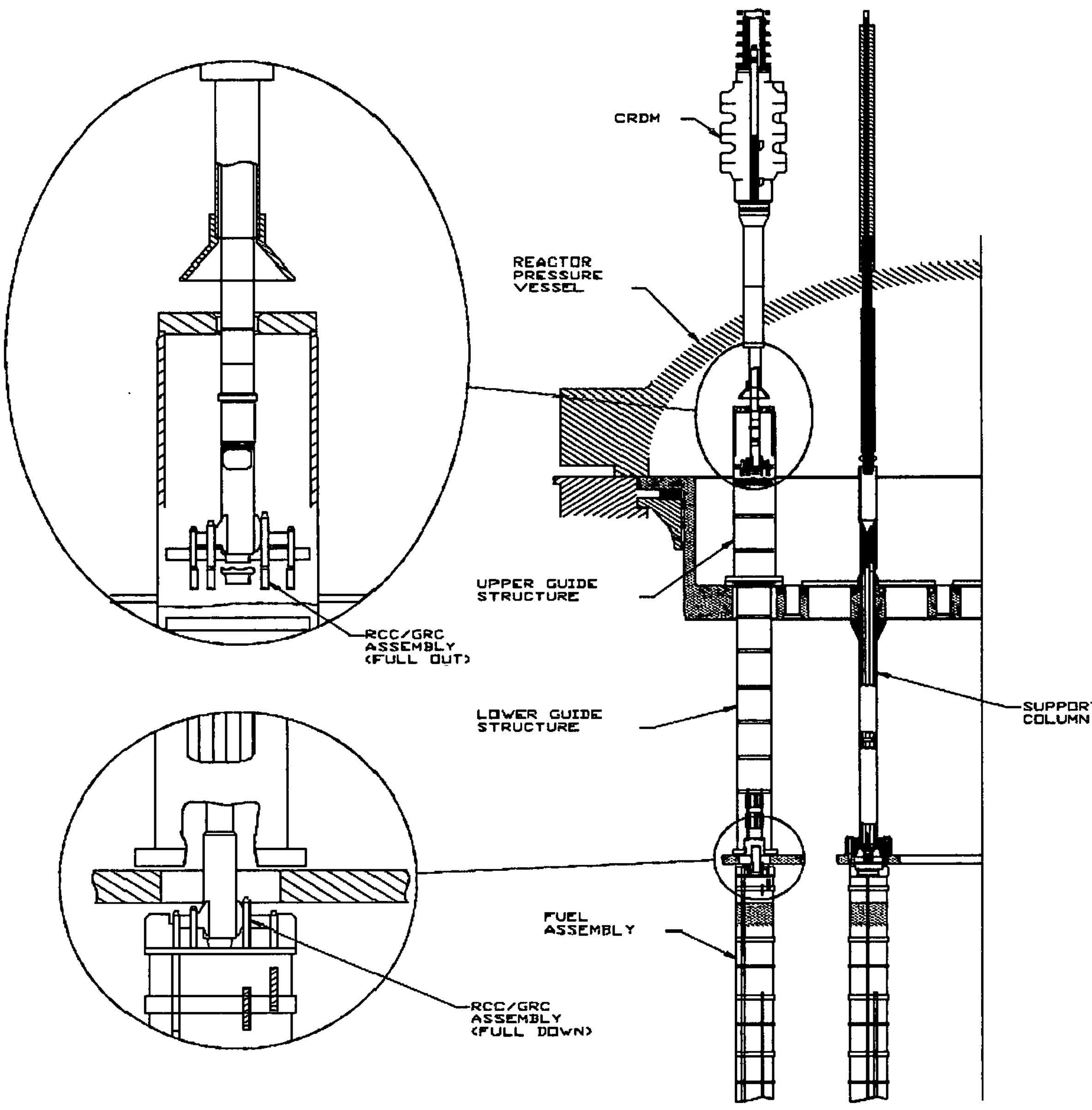


Figure 4.2-8

Rod Cluster Control and Drive Rod Assembly With Interfacing Components

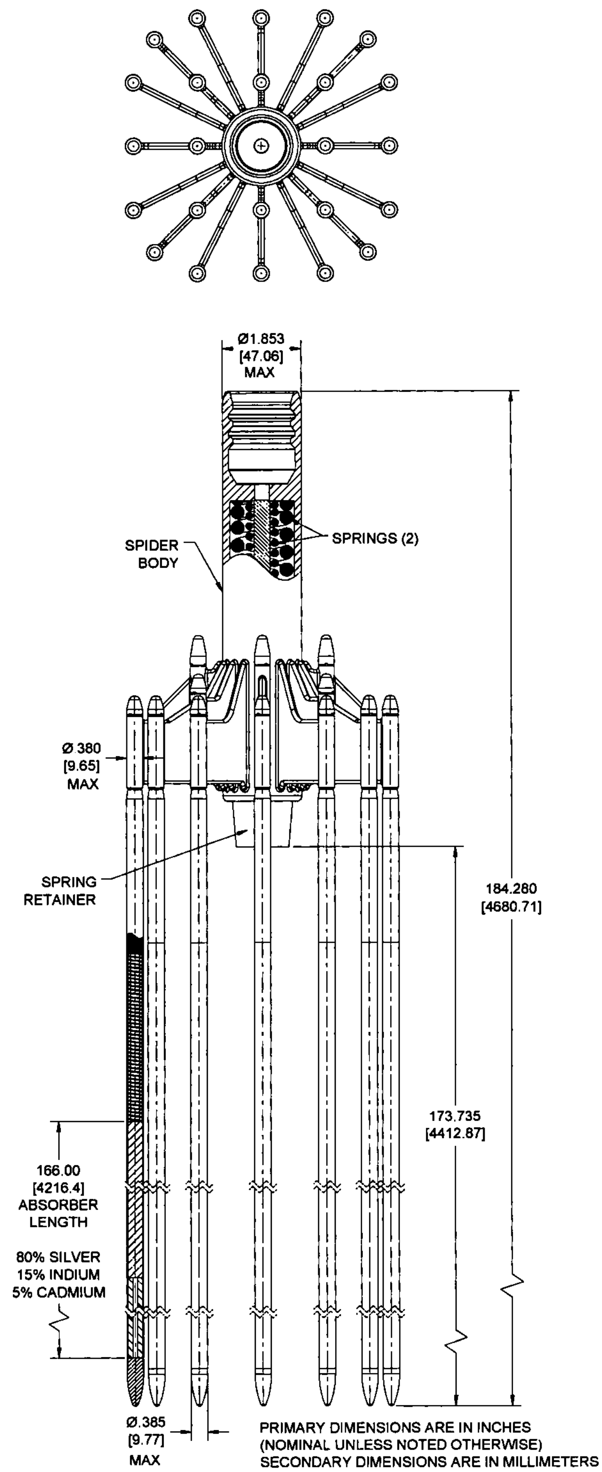
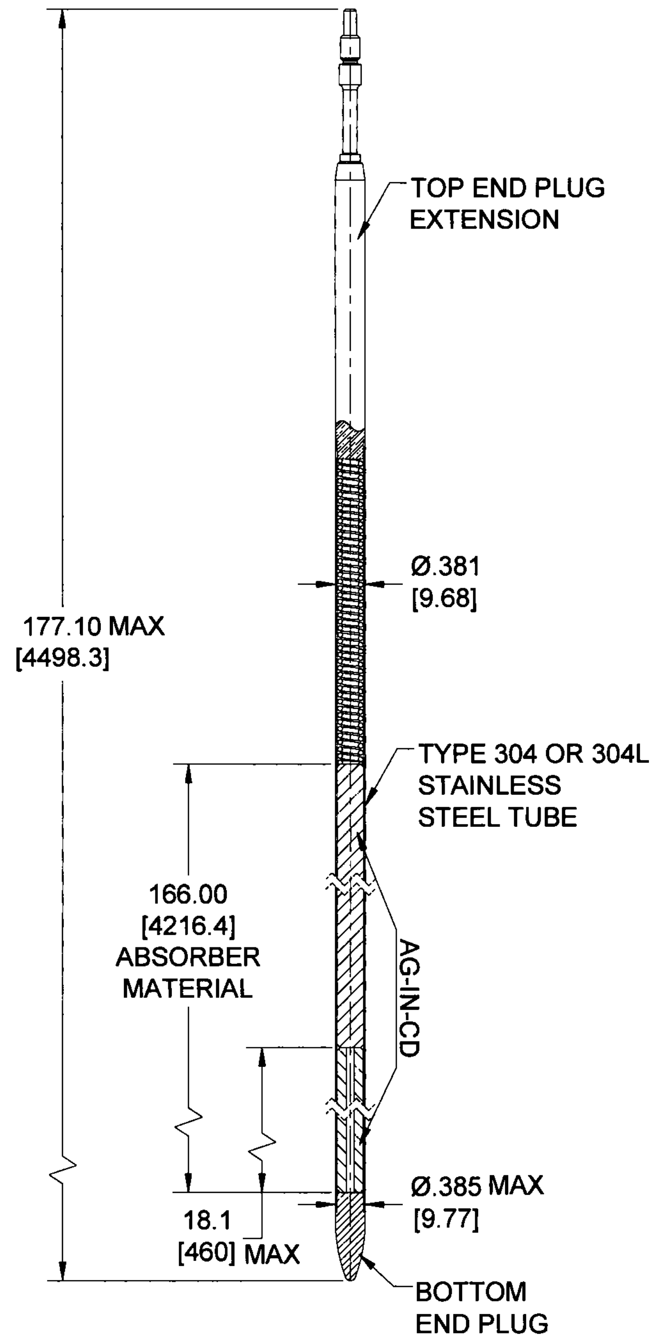


Figure 4.2-9

Rod Cluster Control Assembly



PRIMARY DIMENSIONS ARE IN INCHES
(NOMINAL UNLESS NOTED OTHERWISE)
SECONDARY DIMENSIONS ARE IN MILLIMETERS

Figure 4.2-10

Absorber Rod Detail

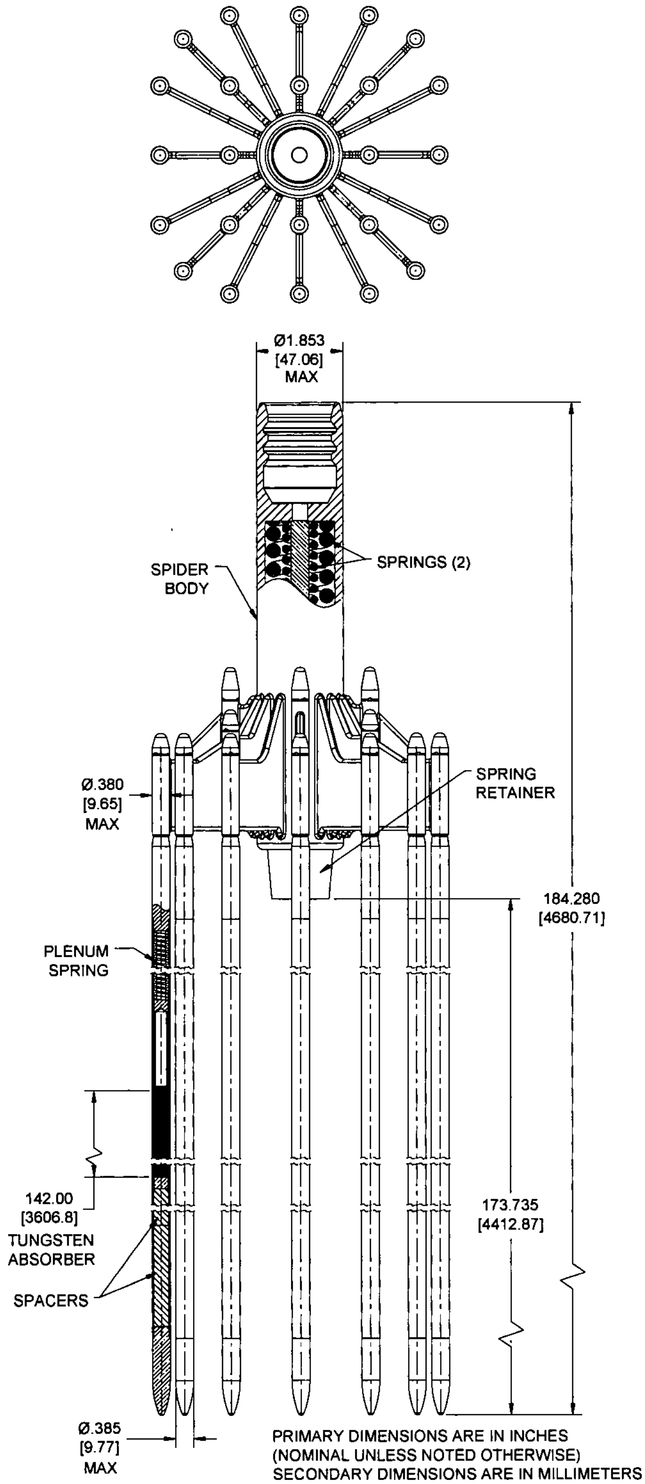


Figure 4.2-11

Gray Rod Cluster Assembly

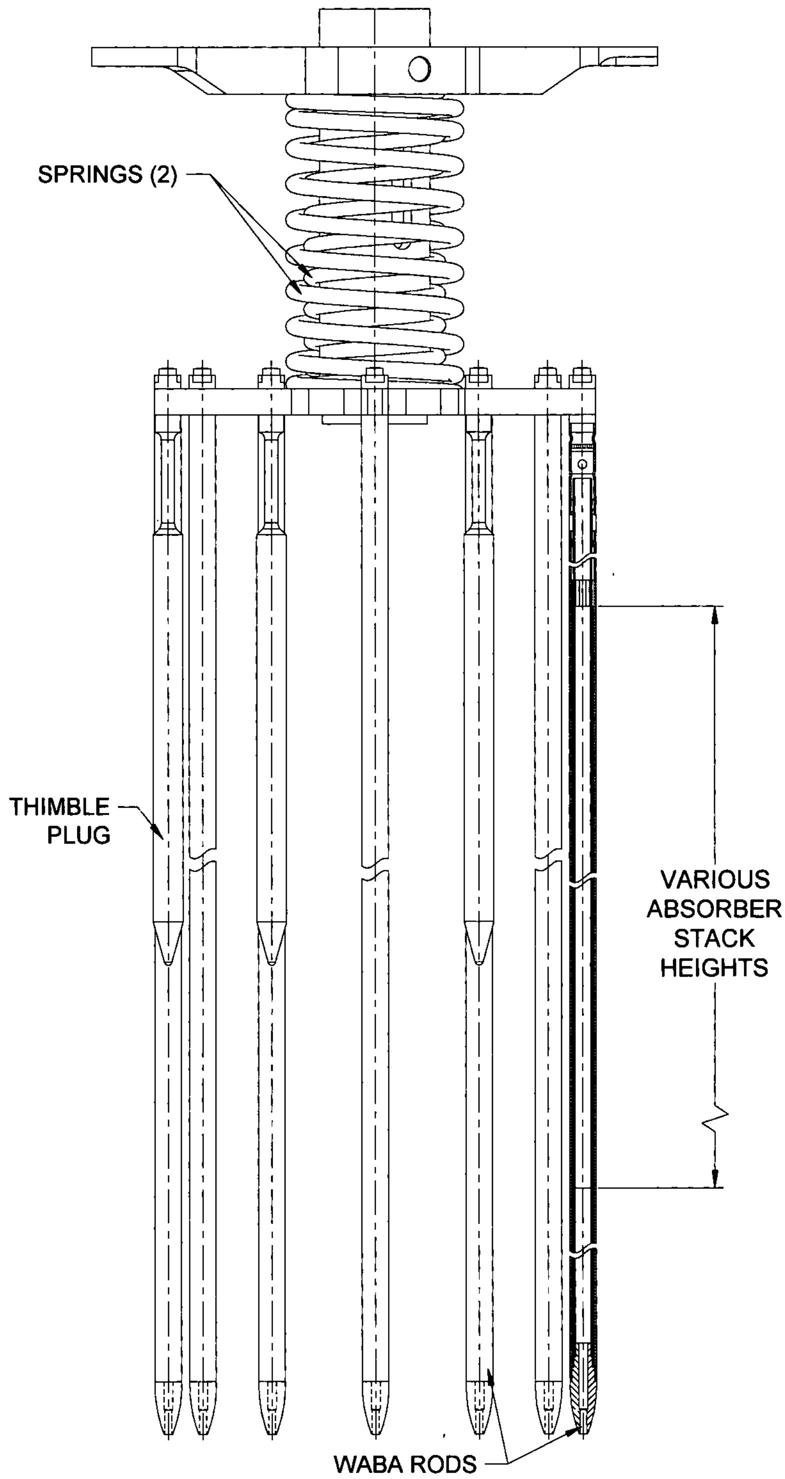


Figure 4.2-12

Wet Annular Burnable Absorber Assembly

Figure 4.2-13

Not used.

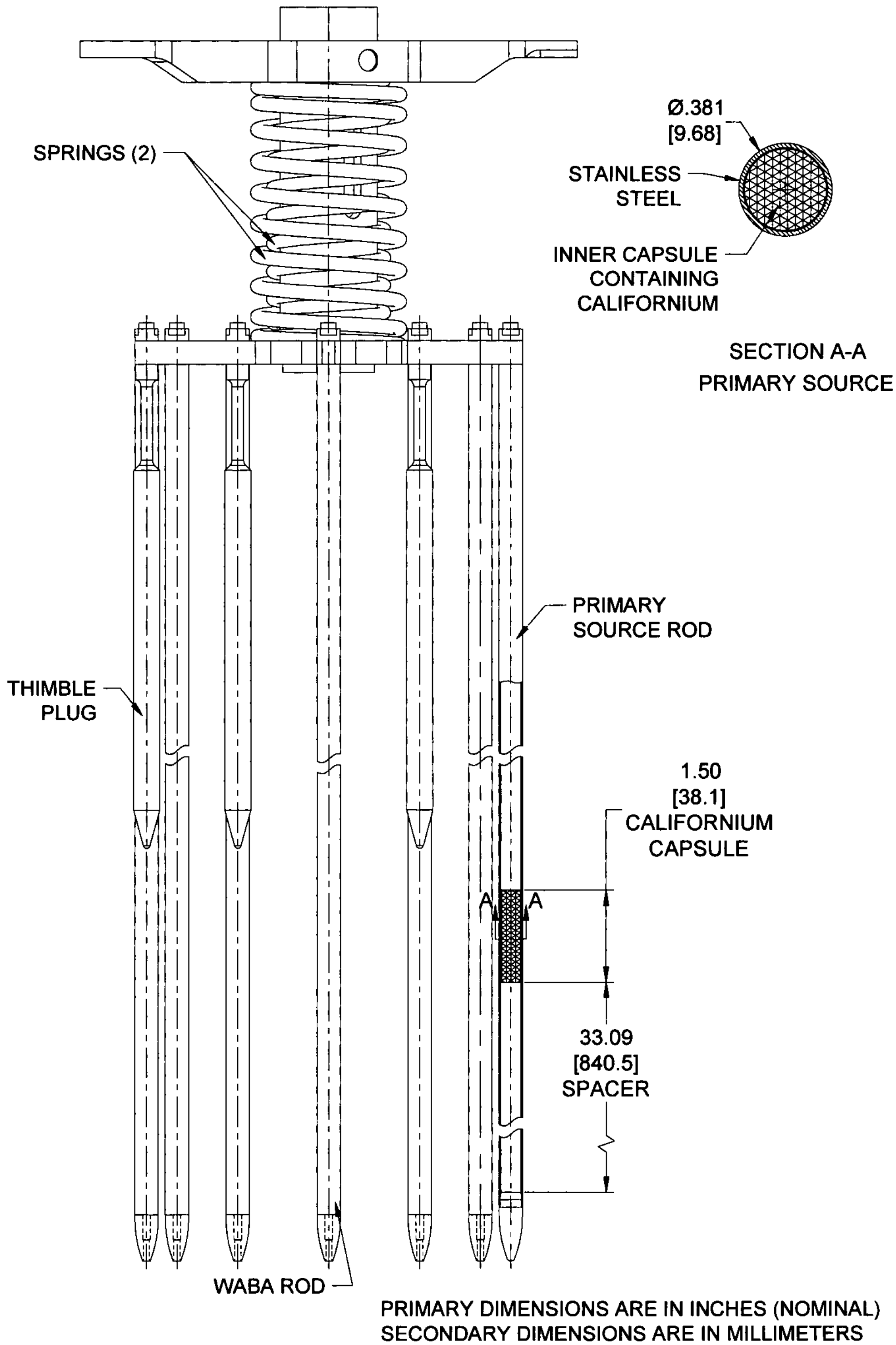


Figure 4.2-14

Primary Source Assembly

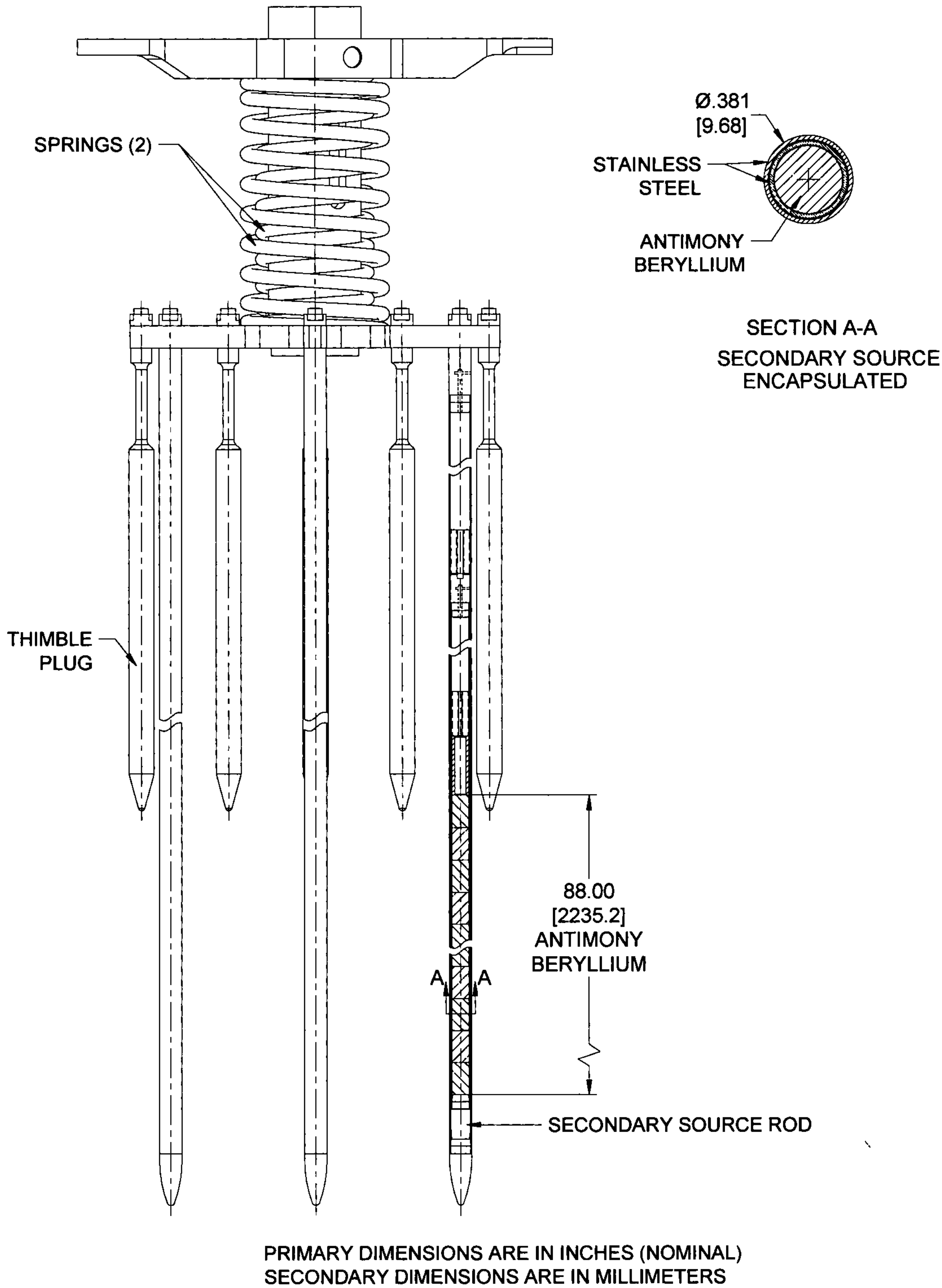


Figure 4.2-15

Secondary Source Assembly

4.3 Nuclear Design

4.3.1 Design Basis

This section describes the design bases and functional requirements used in the nuclear design of the fuel and reactivity control system and relates these design bases to the General Design Criteria (GDC). The design bases are the fundamental criteria that must be met using approved analytical techniques. [*Enhancements to these techniques may be made provided that the changes are founded by NRC approved methodologies as discussed in*]* WCAP-9272-P-A (Reference 1) and [*WCAP-12488-P-A (Reference 2).*]*

The plant conditions for design are divided into four categories:

- Condition I - Normal operation and operational transients
- Condition II - Events of moderate frequency
- Condition III - Infrequent incidents
- Condition IV - Limiting faults

The reactor is designed so that its components meet the following performance and safety criteria:

- In general, Condition I occurrences are accommodated with margin between any plant parameter and the value of that parameter which would require either automatic or manual protective action.

- Condition II occurrences are accommodated with, at most, a shutdown of the reactor with the plant capable of returning to operation after corrective action.

- Fuel damage, that is, breach of fuel rod clad pressure boundary, is not expected during Condition I and Condition II occurrences. A very small amount of fuel damage may occur. This is within the capability of the chemical and volume control system (CVS) and is consistent with the plant design basis.

- Condition III occurrences do not cause more than a small fraction of the fuel elements in the reactor to be damaged, although sufficient fuel element damage might occur to preclude immediate resumption of operation.

- The release of radioactive material due to Condition III occurrences is not sufficient to interrupt or restrict public use of those areas beyond the exclusion area boundary.

- A Condition III occurrence does not by itself generate a Condition IV occurrence or result in a consequential loss of function of the reactor coolant or reactor containment barriers.

- Condition IV faults do not cause a release of radioactive material that results in exceeding the dose limits identified in Chapter 15. Condition IV occurrences are faults that are not expected to occur but are defined as limiting faults which are included in the design.

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

The core design power distribution limits related to fuel integrity are met for Condition I occurrences through conservative design and are maintained by the action of the control system.

The requirements for Condition II occurrences are met by providing an adequate protection system which monitors reactor parameters.

The control and protection systems are described in Chapter 7.

The consequences of Condition II, III, and IV occurrences are described in Chapter 15.

4.3.1.1 Fuel Burnup

4.3.1.1.1 Basis

A limitation on initial installed excess reactivity or average discharge burnup is not required other than as is quantified in terms of other design bases, such as overall negative power reactivity feedback discussed below. [*The NRC has approved, in WCAP-12488-P-A (Reference 2), maximum fuel rod average burnup of 60,000 MWD/MTU. Extended burnup to 62,000 MWD/MTU has been established in Reference 61.*]*

4.3.1.1.2 Discussion

Fuel burnup is a measure of fuel depletion which represents the integrated energy output of the fuel in megawatt-days per metric ton of uranium (MWD/MTU) and is a useful means for quantifying fuel exposure criteria.

The core design lifetime, or design discharge burnup, is achieved by installing sufficient initial excess reactivity in each fuel region and by following a fuel replacement program (such as that described in subsection 4.3.2) that meets the safety-related criteria in each cycle of operation.

Initial excess reactivity installed in the fuel, although not a design basis, must be sufficient to maintain core criticality at full-power operating conditions throughout cycle life with equilibrium xenon, samarium, and other fission products present. Burnable absorbers, control rod insertion, and/or chemical shim are used to compensate for the excess reactivity. The end of design cycle life is defined to occur when the chemical shim concentration is essentially zero with control rods present to the degree necessary for operational requirements. In terms of soluble boron concentration, this corresponds to approximately 10 ppm with the control and gray rods essentially withdrawn.

4.3.1.2 Negative Reactivity Feedbacks (Reactivity Coefficients)

4.3.1.2.1 Basis

For the initial fuel cycle, the fuel temperature coefficient will be negative, and the moderator temperature coefficient of reactivity will be negative for power operating conditions, thereby

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

providing negative reactivity feedback characteristics. The design basis meets General Design Criterion 11.

4.3.1.2.2 Discussion

When compensation for a rapid increase in reactivity is considered, there are two major effects. These are the resonance absorption (Doppler) effects associated with changing fuel temperature and the neutron spectrum and reactor composition change effects resulting from changing moderator density. These basic physics characteristics are often identified by reactivity coefficients. The use of slightly enriched uranium results in a Doppler coefficient of reactivity that is negative. This coefficient provides the most rapid reactivity compensation. The initial core is also designed to have an overall negative moderator temperature coefficient of reactivity during power operation so that average coolant temperature changes or void content provides another, slower compensatory effect. For some core designs, if the compensation for excess reactivity is provided only by chemical shim, the moderator temperature coefficient could become positive. Nominal power operation is permitted only in a range of overall negative moderator temperature coefficient. The negative moderator temperature coefficient can be achieved through the use of discrete burnable absorbers (BAs) and/or integral fuel burnable absorbers and/or control rods by limiting the reactivity controlled by soluble boron.

Burnable absorber content (quantity and distribution) is not stated as a design basis. However, for some reloads, the use of burnable absorbers may be necessary for power distribution control and/or to achieve an acceptable moderator temperature coefficient throughout core life. The required burnable absorber loading is that which is required to meet design criteria.

4.3.1.3 Control of Power Distribution

4.3.1.3.1 Basis

The nuclear design basis is that, with at least a 95 percent confidence level:

- The fuel will not operate with a power distribution that would result in exceeding the departure from nucleate boiling (DNB) design basis (i.e., the departure from nucleate boiling ratio (DNBR) shall be greater than the design limit departure from nucleate boiling ratio as discussed in subsection 4.4.1) under Condition I and II occurrences, including the maximum overpower condition.

- Under abnormal conditions, including the maximum overpower condition, the peak linear heat rate (PLHR) will not cause fuel melting, as defined in subsection 4.4.1.2.

- Fuel management will be such as to produce values of fuel rod power and burnup consistent with the assumptions in the fuel rod mechanical integrity analysis of Section 4.2.

- The fuel will not be operated at Peak Linear Heat Rate (PLHR) values greater than those found to be acceptable within the body of the safety analysis under normal operating

conditions, including an allowance of one percent for calorimetric error (calorimetric uncertainty calculation will be provided per subsection 15.0.15.1).

The above basis meets General Design Criterion 10.

4.3.1.3.2 Discussion

Calculation of extreme power shapes which affect fuel design limits are performed with proven methods. The conditions under which limiting power shapes are assumed to occur are chosen conservatively with regard to any permissible operating state. Even though there is close agreement between calculated peak power and measurements, a nuclear uncertainty is applied (subsection 4.3.2.2.1) to calculated power distribution. Such margins are provided both for the analysis for normal operating states and for anticipated transients.

4.3.1.4 Maximum Controlled Reactivity Insertion Rate

4.3.1.4.1 Basis

The maximum reactivity insertion rate due to withdrawal of rod cluster control assemblies (RCCAs) or gray rod cluster assemblies (GRCAs) or by boron dilution is limited by plant design, hardware, and basic physics. During normal power operation, the maximum controlled reactivity insertion rate is limited. The maximum reactivity change rate for accidental withdrawal of two control banks is set such that PLHR and the departure from nucleate boiling ratio limitations are not challenged. This satisfies General Design Criterion 25.

The maximum reactivity worth of control rods and the maximum rates of reactivity insertion employing control rods are limited 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 an ejection accident. (See Chapter 15).

Following any Condition IV occurrence, such as rod ejection or steam line break, the reactor can be brought to the shutdown condition, and the core maintains acceptable heat transfer geometry. This satisfies General Design Criterion 28.

4.3.1.4.2 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). For this reactor, the maximum control and gray rod speed is 45 inches per minute.

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

4.3.1.5 Shutdown Margins

4.3.1.5.1 Basis

Minimum shutdown margin as specified in the technical specifications is required in all operating modes.

In analyses 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 General Design Criterion 26.

4.3.1.5.2 Discussion

Two independent reactivity control systems are provided: control rods and soluble boron in the coolant. The control rods provide reactivity changes which compensate for the reactivity effects of the fuel and water density changes accompanying power level changes over the range from full load to no load. The control rods provide the minimum shutdown margin under Condition I occurrences and are capable of making the core subcritical rapidly enough to prevent exceeding acceptable fuel damage limits (very small number of rod failures), assuming that the highest worth control rod is stuck out upon trip.

The boron system can compensate for xenon burnout reactivity changes and maintain the reactor in the cold shutdown condition. Thus, backup and emergency shutdown provisions are provided by mechanical and chemical shim control systems which satisfy General Design Criterion 26. Reactivity changes due to fuel depletion are accommodated with the boron system.

4.3.1.5.3 Basis

When fuel assemblies are in the pressure vessel and the vessel head is not in place, k_{eff} will be maintained at or below 0.95 with control rods and soluble boron. Further, the fuel will be maintained sufficiently subcritical that removal of the rod cluster control assemblies will not result in criticality.

4.3.1.5.4 Discussion

ANSI N18.2 (Reference 3) specifies a k_{eff} not to exceed 0.95 in spent fuel storage racks and transfer equipment flooded with pure water and a k_{eff} not to exceed 0.98 in normally dry new fuel storage racks, assuming optimum moderation. No criterion is given for the refueling operation. However, a five percent margin, which is consistent with spent fuel storage and transfer and the new fuel storage, is adequate for the controlled and continuously monitored operations involved.

The boron concentration required to meet the refueling shutdown criteria is specified in the Core Operating Limits Report (COLR). Verification that these shutdown criteria are met, including uncertainties, is achieved using standard design methods. The subcriticality of the core is continuously monitored as described in the technical specifications.

4.3.1.6 Stability

4.3.1.6.1 Basis

The core will be inherently stable to power oscillations at the fundamental mode. This satisfies General Design Criterion 12.

Spatial power oscillations within the core with a constant core power output, should they occur, can be reliably and readily detected and suppressed.

4.3.1.6.2 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; a reactor trip occurs if power increases unacceptably, thereby preserving the design margins to fuel design limits. The combined stability of the turbine, steam generator and the reactor power control systems are such that total core power oscillations are not normally possible. The redundancy of the protection circuits results in a 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; 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.

Indications of power distribution anomalies are continuously available from an online core monitoring system. The online monitoring system processes information provided by the fixed in-core detectors, in-core thermocouples, and loop temperature measurements. Radial power distributions are therefore continuously monitored, thus power oscillations are readily observable and alarmed. The ex-core long ion chambers also provide surveillance and alarms of anomalous power distributions. In proposed core designs, these horizontal plane oscillations are self-damping by virtue of reactivity feedback effects inherent to the basic core physics.

Axial xenon spatial power oscillations may occur during core life, especially late in the cycle. The online core monitoring system provides continuous surveillance of the axial power distributions. The control rod system provides both manual and automatic control systems for controlling the axial power distributions.

Confidence that fuel design limits are not exceeded is provided by reactor protection system overpower ΔT (OPΔT) and overtemperature ΔT (OTΔT) trip functions, which use the loop temperature sensors, pressurizer pressure indication, and measured axial offset as an input. Detection and suppression of xenon oscillations are discussed in subsection 4.3.2.7.

4.3.1.7 Anticipated Transients Without Scram (ATWS)

The AP1000 diverse reactor trip actuation system is independent of the reactor trip breakers used by the protection monitoring system. The diverse reactor trip reduces the probability and consequences of a postulated ATWS. The effects of anticipated transients with failure to trip are not considered in the design bases of the plant. Analysis has shown that the likelihood of such a hypothetical event is negligibly small. Furthermore, analysis of the consequences of a hypothetical failure to trip following anticipated transients has shown that no significant core damage would result, system peak pressures should be limited to acceptable values, and no failure of the reactor coolant system would result. (See WCAP-8330, Reference 5). The process used to evaluate the ATWS risk in compliance with 10 CFR 50.62 is described in Section 15.8 of this DCD.

4.3.2 Description

4.3.2.1 Nuclear Design Description

The reactor core consists of a specified number of fuel rods held in bundles by spacer grids and top and bottom fittings. The fuel rods are fabricated from cylindrical tubes made of zirconium based alloy(s) containing uranium dioxide fuel pellets. The bundles, known as fuel assemblies, are arranged in a pattern which approximates a right circular cylinder.

Each fuel assembly contains a 17 x 17 rod array composed nominally of 264 fuel rods, 24 rod cluster control thimbles, and an in-core instrumentation thimble. Figure 4.2-1 shows a cross-sectional view of a 17 x 17 fuel assembly and the related rod cluster control guide thimble locations. Detailed descriptions of the AP1000 fuel assembly design features are given in Section 4.2.

Both the initial and reload core loading patterns can employ various fuel management techniques including “low-leakage” designs, and are anticipated to operate approximately 18 months between refueling. For reload core loading patterns, the initial and final positions of assemblies, and the number of fresh assemblies and their placement are dependent on the energy requirement for the reload cycle and burnup and power histories of the previous cycles.

For the initial core loading, the fuel rods within certain assemblies contain varying uranium enrichments in both the radial and axial planes. Fuel containing up to five average enrichments will be used in the initial core load to establish a favorable radial power distribution simulating the reactivity distribution of a low leakage reload core. Figure 4.3-1 shows the fuel loading pattern used in the initial cycle. The higher enriched regions will be configured in the core interior consistent with the feed fuel placement in a reload core, and the lower enriched regions will approximate the reactivity of the burned fuel assemblies of a reload core. The enrichments for the initial cycle are shown in Table 4.3-1.

The core average enrichment is determined by the amount of fissionable material required to provide the desired energy requirements. 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. In addition, the fission process

results in the formation of fission products, some of which readily absorb neutrons. These effects, the depletion and the buildup of fission products, are partially offset by the buildup of plutonium shown in Figure 4.3-2 for a typical 17 x 17 fuel assembly, which occurs due to the parasitic absorption of neutrons in U-238. Therefore, at the beginning of any cycle a reactivity reserve equal to the depletion of the fissionable fuel and the buildup of fission product poisons less the buildup of fissile fuel over the specified cycle life is built into the reactor. This excess reactivity is controlled by removable neutron-absorbing material in the form of boron dissolved in the primary coolant, control rod insertion, burnable absorber rods, and integral fuel burnable absorbers (IFBA). The stack length of the burnable absorber rods and/or integral absorber bearing fuel may vary for different core designs, with the optimum length determined on a design specific basis. Figure 4.3-3 is a plot of the core soluble boron concentration versus core depletion for the first operating cycle.

The concentration of the soluble neutron absorber is varied to compensate for reactivity changes due to fuel burnup, fission product poisoning including xenon and samarium, burnable absorber depletion, and the cold-to-operating moderator temperature change. Throughout the operating range, the CVS is designed to provide changes in reactor coolant system (RCS) boron concentration to compensate for the reactivity effects of fuel depletion, peak xenon burnout and decay, and cold shutdown boration requirements.

Burnable absorbers are strategically located to provide a favorable radial power distribution and provide for negative reactivity feedback. Figures 4.3-4a and 4.3-4b show the burnable absorber distributions within a fuel assembly for the several patterns used in a 17 x 17 array. The initial core burnable absorber loading pattern is shown in Figure 4.3-5.

Tables 4.3-1 through 4.3-3 contain summaries of reactor core design parameters including reactivity coefficients, delayed neutron fraction, and neutron lifetimes. Sufficient information is included to permit an independent calculation of the nuclear performance characteristics of the core.

4.3.2.2 Power Distribution

The accuracy of power distribution calculations has been confirmed through approximately 1000 flux maps under conditions very similar to those expected. Details of this confirmation are given in WCAP-7308-L-P-A (Reference 7) and in subsection 4.3.2.2.7.

4.3.2.2.1 Definitions

Relative power distributions within the reactor are quantified in terms of hot channel factors. These hot channel factors are normalized ratios of maximal absolute power generation rates and are a measure of the peak pellet power within the reactor core relative to the average pellet (F_Q) and the energy produced in a coolant channel relative to the core average channel ($F_{\Delta H}$). Absolute power generation rates are expressed in terms of quantities related to the nuclear or thermal design; more specifically, volumetric power density (q_{vol}) is the thermal power produced per unit volume of the core (kW/liter).

Linear heat rate (LHR) is the thermal power produced per unit length of active fuel (kW/ft). Since fuel assembly geometry is standardized, LHR is the unit of absolute power density

most commonly used. For practical purposes, LHR differs from q_{vol} by a constant factor which includes geometry effects and the heat flux deposition fraction. The peak linear heat rate (PLHR) is defined as the maximum linear heat rate occurring throughout the reactor. PLHR directly impacts fuel temperatures and decay power levels thus being a significant safety analysis parameter.

Average linear heat rate (ALHR) is the total thermal power produced in the fuel rods expressed as heat flux divided by the total active fuel length of the rods in the core.

Local heat flux is the heat flux at the surface of the cladding (Btu/hr-ft^2). For nominal rod parameters, this differs from linear heat rate by a constant factor.

Rod power is the total power generated in one rod (kW).

Average rod power is the total thermal power produced in the fuel rods divided by the number of fuel rods (assuming the rods have equal length).

The hot channel factors used in the discussion of power distributions in this section are defined as follows:

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 rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods.

F^N_Q, **nuclear heat flux hot channel factor,** is defined as the maximum local fuel rod linear heat rate divided by the average fuel rod linear heat rate, assuming nominal fuel pellet and rod parameters.

F^E_Q, **engineering heat flux hot channel factor,** is the allowance on heat flux required for manufacturing tolerances. The engineering factor allows for local variations in enrichment, pellet density and diameter, burnable absorber content, surface area of the fuel rod, and eccentricity of the gap between pellet and clad. Combined statistically, the net effect is a factor of 1.03 to be applied to the fuel rod surface heat flux.

$F^N_{\Delta H}$, **nuclear enthalpy rise hot channel factor,** is defined as the ratio of the maximum integrated rod power within the core to the average rod power.

Manufacturing tolerances, hot channel power distribution, and surrounding channel power distributions are treated explicitly in the calculation of the departure from nucleate boiling ratio described in Section 4.4.

It is convenient for the purposes of discussion to define subfactors of F_Q. However, design limits are set in terms of the total peaking factor.

F_Q = total peaking factor or heat flux hot channel factor = $\dfrac{PLHR}{ALHR}$

Without densification effects:

$$F_Q = F_Q^N \times F_Q^E = F_{XY}^N \times F_Z^N \times F_U^N \times F_Q^E$$

where F^N_Q and F^E_Q are defined above and:

F^N_U = factor for calculational uncertainty, assumed to be 1.05.

F^N_{XY} = ratio of peak power density to average power density in the horizontal plane of peak local power.

F^N_Z = ratio of the power per unit core height in the horizontal plane of peak local power to the average value of power per unit core height. If the plane of peak local power coincides with the plane of maximum power per unit core height, then F^N_Z is the core average axial peaking factor.

4.3.2.2.2 Radial Power Distributions

The power shape in horizontal sections of the core at full power is a function of the fuel assembly and burnable absorber loading patterns, the control rod pattern, and the fuel burnup distribution. Thus, at any time in the cycle, a horizontal section of the core can be characterized as unrodded or with control rods. These two situations combined with burnup effects determine the radial power shapes which can exist in the core at full power. Typical first cycle values of $F^N_{\Delta H}$, the nuclear enthalpy rise hot channel factors from beginning of life (BOL) to end of life (EOL) are given in Table 4.3-2. The effects on radial power shapes of power level, xenon, samarium, and moderator density effects are also considered, but these are quite small. The effect of nonuniform flow distribution is negligible. While radial power distributions in various planes of the core are often illustrated, since the moderator density is directly proportional to enthalpy, the core radial enthalpy rise distribution, as determined by the integral of power up each channel, is of greater interest. Figures 4.3-6 through 4.3-11 show typical normalized power density distributions for one-quarter of the core for representative operating conditions. These conditions are as follows:

- Hot full power (HFP) near beginning of life, unrodded, no xenon
- Hot full power near beginning of life, unrodded, equilibrium xenon
- Hot full power near beginning of life, gray bank MA+MB in, equilibrium xenon
- Hot full power near middle of life (MOL), unrodded equilibrium xenon
- Hot full power near end of life, unrodded, equilibrium xenon
- Hot full power near end of life, gray bank MA+MB in, equilibrium xenon

Since the position of the hot channel varies from time to time, a single-reference radial design power distribution is selected for departure from nucleate boiling calculations. This reference

power distribution is chosen conservatively to concentrate power in one area of the core, minimizing the benefits of flow redistribution. Assembly powers are normalized to core average power. The radial power distribution within a fuel rod and its variation with burnup as utilized in thermal calculations and fuel rod design are discussed in Section 4.4.

4.3.2.2.3 Assembly Power Distributions

For the purpose of illustration, typical rodwise power distributions from the beginning of life and end of life conditions corresponding to Figures 4.3-7 and 4.3-10, are given for representative hot channel assemblies in Figures 4.3-12 and 4.3-13, respectively.

Since the detailed power distribution surrounding the hot channel varies from time to time, a conservatively flat radial assembly power distribution is assumed in the departure from nucleate boiling analysis, described in Section 4.4, with the rod of maximum integrated power artificially raised to the design value of $F^N_{\Delta H}$. Care is taken in the nuclear design of the fuel cycles and operating conditions to confirm that a flatter assembly power distribution does not occur with limiting values of $F^N_{\Delta H}$.

4.3.2.2.4 Axial Power Distributions

The distribution of power in the axial or vertical direction is largely under the control of the operator through either the manual operation of the control rods or the automatic motion of control rods in conjunction with manual operation of the chemical and volume control system. The automated mode of operation is referred to as mechanical shim (MSHIM) and is discussed in subsection 4.3.2.4.16. The rod control system automatically modulates the insertion of the axial offset (AO) control bank controlling the axial power distribution simultaneous with the MSHIM gray and control rod banks to maintain programmed coolant temperature. Operation of the chemical and volume control system is initiated manually by the operator to compensate for fuel burnup and maintain the desired MSHIM bank insertion. Nuclear effects which cause variations in the axial power shape include moderator density, Doppler effect on resonance absorption, spatial distribution of xenon, burnup, and axial distribution of fuel enrichment and burnable absorber. Automatically controlled variations in total power output and rod motion are also important in determining the axial power shape at any time.

The online core monitoring system provides the operator with detailed power distribution information in both the radial and axial sense continuously using signals from the fixed in-core detectors. Signals are also available to the operator from the ex-core ion chambers, which are long ion chambers outside the reactor vessel running parallel to the axis of the core. Separate signals are taken from each ion chamber. The ion chamber signals are processed and calibrated against in-core measurements such that an indication of the power in the top of the core less the power in the bottom of the core is derived. The calibrated difference in power between the core top and bottom halves, called the flux difference (ΔI), is derived for each of the four channels of ex-core detectors and is displayed on the control panel. The principal use of the flux difference is to provide the shape penalty function to the OTΔT DNB protection and the OPΔT overpower protection.

4.3.2.2.5 Local Power Peaking

Fuel densification occurred early in the evolution of pressurized water reactor fuel manufacture under irradiation in several operating reactors. This caused the fuel pellets to shrink both axially and radially. The pellet shrinkage combined with random hang-up of fuel pellets can result in gaps in the fuel column when the pellets below the hung-up pellet settle in the fuel rod. The gaps vary in length and location in the fuel rod. Because of decreased neutron absorption in the vicinity of the gap, power peaking occurs in the adjacent fuel rods, resulting in an increased power peaking factor. A quantitative measure of this local peaking is given by the power spike factor S(Z), where Z is the axial location in the core. The power spike factor S(z) is discussed in References 8, 9, and 10.

Modern PWR fuel manufacturing practices have essentially eliminated significant fuel densification impacts on reactor design and operation. It has since been concluded and accepted that a densification power spike factor of 1.0 is appropriate for Westinghouse fuel as described in WCAP-13589-A (Reference 59).

4.3.2.2.6 Limiting Power Distributions

According to the ANSI classification of plant conditions (Chapter 15), Condition I occurrences are those expected frequently or regularly in the course of power operation, maintenance, or maneuvering of the plant. As such, Condition I occurrences are accommodated with margin between any plant parameter and the value of that parameter which would require either automatic or manual protective action. Condition I occurrences are considered from the point of view of affecting the consequences of fault conditions (Conditions II, III, and IV). Analysis of each fault condition described is based on a conservative set of corresponding initial conditions.

The list of steady-state and shutdown conditions, permissible deviations, and operational transients is given in Chapter 15. Implicit in the definition of normal operation is proper and timely action by the reactor operator; that is, the operator follows recommended operating procedures for maintaining appropriate power distributions and takes any necessary remedial actions when alerted to do so by the plant instrumentation.

The online monitoring system evaluates the consequences of limiting power distributions based upon the conditions prevalent in the reactor at the current time. Operating space evaluations performed by the online monitoring system include the most limiting power distributions that can be generated by inappropriate operator or control system actions given the current core power level, xenon distribution, MSHIM or AO bank insertion and core burnup. Thus, as stated, the worst or limiting power distribution which can occur during normal operation is considered as the starting point for analysis of Conditions II, III, and IV occurrences.

Improper procedural actions or errors by the operator are assumed in the design as occurrences of moderate frequency (Condition II). Some of the consequences which might result are discussed in Chapter 15. Therefore, the limiting power shapes which result from such Condition II occurrences are those power distributions which deviate from the normal operating condition within the allowable operating space as defined in the core operating

limits; e.g., due to lack of proper action by the operator during a xenon transient following a change in power level brought about by control rod motion. Power distributions which fall in this category are used for determination of the reactor protection system setpoints to maintain margin to overpower or departure from nucleate boiling limits.

The means for maintaining power distributions within the required absolute power generation limits are described in the technical specifications. The online core monitoring system provides the operator with the current allowable operating space, detailed current power distribution information, thermal margin assessment and operational recommendations to manage and maintain required thermal margins. As such, the online monitoring system provides the primary means of managing and maintaining required operating thermal margins during normal operation.

In the unlikely event that the online monitoring system is out of service, power distribution controls based on bounding, precalculated analysis are also provided to the operator such that the online monitoring system is not a required element for short term reactor operation. Limits are placed on the axial flux difference so that the heat flux hot channel factor F_Q is maintained within acceptable limits. A discussion of precalculated power distribution control in Westinghouse pressurized water reactors (PWRs) is included in WCAP-7811 (Reference 11). 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 WCAP-8385 (Reference 12). 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 $F_{\Delta H}^N$, include the nuclear effects which influence the radial and axial power distributions throughout core life for various modes of operation, including load follow, reduced power operation, and axial xenon transients.

Power distributions are calculated for the full-power condition. Fuel and moderator temperature feedback effects are included within these calculations in each spatial dimension. The steady-state nuclear design calculations are done for normal flow with the same mass flow in each channel and flow redistribution effects neglected. The effect of flow redistribution is calculated explicitly where it is important in the departure from nucleate boiling 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.

The core axial profile 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 thermal margin 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 significant to the axial power distribution analysis are as follows:

- Core power level

- Core height
- Coolant temperature and flow
- Coolant temperature program as a function of reactor power
- Fuel cycle lifetimes
- Rod bank worth
- Rod bank overlaps

Normal operation of the plant assumes compliance with the following conditions:

- Control rods in a single bank move together with no individual rod insertion differing from the bank demand position by more than the number of steps identified in the technical specifications.

- Control banks are sequenced with overlapping banks.

- The control bank insertion limits are not violated.

- Axial power distribution control procedures, which are given in terms of flux difference control and control bank position, are observed.

The axial power distribution procedures referred to above are part of the required operating procedures followed in normal operation with the online monitoring system out of service. In service, the online core monitoring system provides continuous indication of power distribution, shutdown margin, and margin to design limits.

The relaxed axial offset control (RAOC) procedures described in WCAP-10216-P-A (Reference 13) were developed to provide wide control band widths and consequently, more operating flexibility. These wide operating limits, particularly at lower power levels, increase plant availability by allowing quicker plant startup and increased maneuvering flexibility without trip. This procedure has been modified to accommodate AP1000 MSHIM operation. It is applied to analysis of axial power distributions under MSHIM control for the purpose of defining the allowed normal operating space such that Condition I thermal margin limits are maintained and Condition II occurrences are adequately protected by the reactor protection system when the online monitoring system is out of service.

The purpose of this analysis is to find the widest permissible ΔI versus power operating space by analyzing a wide range of achievable xenon distributions, MSHIM/AO bank insertion, and power level.

The bounding analyses performed off line in anticipation of the online monitoring system being out of service is similar to that based on the relaxed axial offset control analysis, which uses a xenon reconstruction model described in WCAP-10216-P-A (Reference 13). This is a practical method which is used to define the power operating space allowed with AP1000 MSHIM operation. Each resulting power shape is analyzed to determine if loss-of-coolant accident constraints are met or exceeded.

The online monitoring system evaluates the effects of radial xenon distribution changes due to operational parameter changes continuously and therefore eliminates the need for overly

conservative bounding evaluations when the online monitoring system is available. A detailed discussion of this effect may be found in WCAP-8385 (Reference 12). The calculated values have been increased by a factor of 1.05 for method uncertainty and a factor of 1.03 for the engineering factor F^E_Q.

The envelope drawn in Figure 4.3-14 represents an upper bound envelope on local power density versus elevation in the core. This envelope is a conservative representation of the bounding values of local power density.

The online monitoring system measures the core condition continuously and evaluates the thermal margin condition directly in terms of peak linear heat rate and margin to departure from nucleate boiling limitations directly.

Allowing for fuel densification effects, the average linear power at 3400 MW is 5.72 kW/ft. From Figure 4.3-14, the conservative upper bound value of normalized local power density, including uncertainty allowances, is 2.60 corresponding to a peak linear heat rate of 15.0 kW/ft at each core elevation at 101 percent power.

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

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

The second category assumes that the operator mispositions the AO and/or MSHIM rod banks 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 power distribution limit violation (such as boration/dilution transient) assuming automatic operation of the rod control system which will maintain constant reactor power.

For each of the above categories, the trip setpoints are designed so as not to exceed fuel centerline melt criteria as well as fuel mechanical design criteria.

The appropriate hot channel factors F_Q and $F_{\Delta H}^N$ for peak local power density and for DNB analysis at full power are based on analyses of possible operating power shapes and are addressed in the technical specifications.

The maximum allowable F_Q can be increased with decreasing power, as shown in the technical specifications. Increasing $F_{\Delta H}^N$ 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 subsection 4.4.4.3. The allowance for increased $F_{\Delta H}^N$ permitted is addressed in the technical specifications.

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 $F_{\Delta H}^N$ for possible rod configurations occurring in normal operation are used in verifying that this criterion is met. The worst values generally occur when the rods are assumed to be at their insertion limits. Operation with rod positions above the allowed rod insertion limits provides increased margin to the $F_{\Delta H}^N$ criterion. As discussed in Section 3.2 of WCAP-7912-P-A (Reference 14), it has been determined that the technical specifications limits are met, provided the above conditions are observed. These limits are taken as input to the thermal-hydraulic design basis, as described in subsection 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 precondition 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 described in Chapter 7.

The independence of the various individual uncertainties constituting the uncertainty factor on F_Q enables the uncertainty (F^U_Q) to be calculated by statistically combining the individual uncertainties on the limiting rod. The standard deviation of the resultant distribution of F^U_Q is determined by taking the square root of the sum of the variances of each of the contributing distributions WCAP-7308-L-P-A (Reference 7). The values for F^E_Q and F^N_U are 1.03 and 1.05, respectively. The value for the rod bow factor, F^B_Q, is 1.056, which accounts for the maximum F_Q penalty as a function of burnup due to rod bow effects.

4.3.2.2.7 Experimental Verification of Power Distribution Analysis

This subject is discussed in WCAP-7308-L-P-A (Reference 7) and WCAP-12472-P-A (Reference 4). A summary of these reports and the extension to include the fixed in-core instrumentation system is given below. Power distribution related measurements are incorporated into the evaluation of calculated power distribution information using the in-core instrumentation processing algorithms contained within the online monitoring system. The processing algorithms contained within the online monitoring system are functionally identical to those historically used for the evaluation of power distribution measurements in

Westinghouse PWRs. Advances in technology allow a complete functional integration of reaction rate measurement algorithms and the expected reaction rate predictive capability within the same software package. The predictive software integrated within the online monitoring system supplies accurate, detailed information of current reactor conditions. The historical algorithms are described in detail in WCAP-12472-P-A (Reference 4).

The measured versus calculational comparison is performed continuously by the online monitoring system throughout the core life. The online monitoring system operability requirements are specified in the technical specifications.

In a measurement of the reactor power distribution and the associated thermal margin limiting parameters, with the in-core instrumentation system described in subsections 7.7.1 and 4.4.6, the following uncertainties must be considered:

A. Reproducibility of the measured signal

B. Errors in the calculated relationship between detector current and local power generation within the fuel bundle

C. Errors in the detector current associated with the depletion of the emitter material, manufacturing tolerances and measured detector depletion

D. Errors due to the inference of power generation some distance from the measurement thimble

The appropriate allowance for category A has been accounted for through the imposition of strict manufacturing tolerances for the individual detectors. This approach is accepted industry practice and has been used in PWRs with fixed in-core instrumentation worldwide. Errors in category B above are quantified by calculation and evaluation of critical experiment data on arrays of rods with simulated guide thimbles, control rods, burnable absorbers, etc. These critical experiments provide the quantification of errors of categories A and D above. Errors in category C have been quantified through direct experimental measurement of the depletion characteristics of the detectors being used including the precision of the in-core instrumentation systems measurement of the current detector depletion. The description of the experimental measurement of detector depletion can be found in EPRI-NP-3814 (Reference 16).

WCAP-7308-L-P-A (Reference 7) describes critical experiments performed at the Westinghouse Reactor Evaluation Center and measurements taken on two Westinghouse plants with movable fission chamber in-core instrumentation systems. The measurement aspects of the movable fission chamber share the previous uncertainty categories less category C which is independent of the other sources of uncertainty. WCAP-7308-L-P-A (Reference 7) concludes that the uncertainty associated with peak linear heat rate (F_Q*P) is less than five percent at the 95 percent confidence level with only five percent of the measurements greater than the inferred value.

In comparing measured power distributions (or detector currents) with calculations for the same operating conditions, it is not possible to isolate the detector reproducibility. Thus, a

comparison between measured and predicted power distributions includes some measurement error. Such a comparison is given in Figure 4.3-15 for one of the maps used in WCAP-7308-L-P-A (Reference 7). Since the first publication of WCAP-7308-L-P-A, hundreds of measurements have been taken on reactors all over the world. These results confirm the adequacy of the five percent uncertainty allowance on the calculated peak linear heat rate (ALHR*F_Q*P).

A similar analysis for the uncertainty in hot rod integrated power $F_{\Delta H}$*P measurements results in an allowance of four percent at the equivalent of a 95 percent confidence level.

A measurement in the fourth cycle of a 157-assembly, 12-foot core is compared with a simplified one-dimensional core average axial calculation in Figure 4.3-16. This calculation does not give explicit representation to the fuel grids.

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

- Much of the data is obtained in steady-state operation at constant power in the normal operating configuration.
- Data with unusual values of axial offset are obtained as part of the ex-core detector calibration exercise performed monthly.
- 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 WCAP-7912-P-A (Reference 14). Figure 4.3-17 contains a summary of measured values of F_Q as a function of axial offset for five plants from that report.

4.3.2.2.8 Testing

A series of physics tests are planned to be performed on the first core. 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 the selected safety-related parameters of the reload design.

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 WCAP-12472-P (Reference 4). The relevant conclusions are summarized in subsection 4.3.2.2.7 and subsection 4.4.6.

Provided the limitations given in subsection 4.3.2.2.6 on rod insertion and flux difference are observed, the in-core and ex-core detector systems provide adequate monitoring of power distributions when the online monitoring system is out of service. Further details of specific

limits on the observed rod positions and flux difference are given in the technical specifications, together with a discussion of their bases.

Limits for alarms and reactor trip are given in 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. These kinetic characteristics are quantified in reactivity coefficients. The reactivity coefficients reflect the changes in the neutron multiplication due to varying plant conditions, such as thermal power, moderator and fuel temperatures, coolant pressure, or void conditions, although the latter are relatively unimportant. 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 with approved nuclear 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 five percent typically changes the moderator and Doppler temperature coefficients by less than 0.01 pcm/°F. An artificially skewed xenon distribution which results in changing the radial $F_{\Delta H}^{N}$ by three percent typically changes the moderator and Doppler temperature coefficients by less than 0.03 pcm/°F and 0.001 pcm/°F, respectively. The spatial effects are accentuated in some transient conditions, for example, in postulated rupture of the main steam line and rupture of a rod cluster control assembly mechanism housing described in subsections 15.1.5 and 15.4.8, and are included in these analyses.

The analytical methods and calculational models used in calculating the reactivity coefficients are given in subsection 4.3.3. These models have been confirmed through extensive qualification efforts performed for core and lattice designs.

Quantitative information for calculated reactivity coefficients including fuel-Doppler coefficient, moderator coefficients (density, temperature, pressure, and void), and power coefficient, is given in the following sections.

4.3.2.3.1 Fuel Temperature (Doppler) Coefficient

The fuel temperature (Doppler) coefficient is defined as the change in reactivity per degree change in effective fuel temperature and is primarily a measure of the Doppler broadening of U-238 and Pu-240 resonance absorption peaks. Doppler broadening of other isotopes is also considered, but their contribution to the Doppler effect is small. An increase in fuel temperature increases the effective resonance absorption cross sections of the fuel and produces a corresponding reduction in reactivity.

The fuel temperature coefficient is calculated using approved nuclear methods. Moderator temperature is held constant, and the power level is varied. Spatial variation of fuel temperature is taken into account by calculating the effective fuel temperature as a function of power density, as discussed in subsection 4.3.3.1.

A typical Doppler temperature coefficient is shown in Figure 4.3-18 as a function of the effective fuel temperature (at beginning of life and end of life conditions). The effective fuel temperature is lower than the volume-averaged fuel temperature, since the neutron flux distribution is non-uniform through the pellet and gives preferential weight to the surface temperature. A typical Doppler-only contribution to the power coefficient, defined later, is shown in Figure 4.3-19 as a function of relative core power. The integral of the differential curve in Figure 4.3-19 is the Doppler contribution to the power defect and is shown in Figure 4.3-20 as a function of relative power. The Doppler temperature coefficient becomes more negative as a function of life as the Pu-240 content increases, thus increasing the Pu-240 resonance absorption. The upper and lower limits of Doppler coefficient used in accident analyses are given in Chapter 15.

4.3.2.3.2 Moderator Coefficients

The moderator coefficient is a measure of the change in reactivity due to a change in specific coolant parameters, such as density/temperature, pressure, or void. The coefficients obtained are moderator density/temperature, pressure, and void coefficients.

4.3.2.3.2.1 Moderator Density and Temperature Coefficients

The moderator temperature (density) coefficient is defined as the change in reactivity per degree change in the moderator temperature. Generally, the effects of the changes in moderator density and the temperature are considered together.

The soluble boron used in the reactor as a means of reactivity control also has an effect on the moderator density coefficient, since the soluble boron density and the water density are decreased when the coolant temperature rises. A decrease in the soluble boron density introduces a positive component in the moderator coefficient. If the concentration of soluble boron is large enough, the net value of the coefficient may be positive.

The initial core hot boron concentration is sufficiently low that the moderator temperature coefficient is negative at operating temperatures with the burnable absorber loading specified. Discrete or integral fuel burnable absorbers can be used in reload cores to confirm the moderator temperature coefficient is negative over the range of power operation. The effect of control rods is to make the moderator coefficient more negative, since the thermal neutron mean free path, and hence the volume affected by the control rods, increase with an increase in temperature.

With burnup, the moderator coefficient becomes more negative, primarily as a result of boric acid dilution, but also to a significant extent from the effects of the buildup of plutonium and fission products.

The moderator coefficient is calculated for a range of plant conditions by performing two group two- or three-dimensional calculations, in which the moderator temperature is varied by about ±5°F about each of the mean temperatures, resulting in density changes consistent with the temperature change. The moderator temperature coefficient is shown as a function of core temperature and boron concentration for the core in Figures 4.3-21 through 4.3-23. The temperature range covered is from cold, about 70°F, to about 550°F. The contribution due to Doppler coefficient (because of change in moderator temperature) has been subtracted from these results. Figure 4.3-24 shows the unrodded, hot full-power moderator temperature coefficient plotted as a function of burnup for the initial cycle. The temperature coefficient corresponds to the unrodded critical boron concentration present at hot full power operating conditions.

The moderator coefficients presented here are calculated to describe the core behavior in normal and accident situations when the moderator temperature changes can be considered to affect the entire core.

4.3.2.3.2.2 Moderator Pressure Coefficient

The moderator pressure coefficient relates the change in moderator density, resulting from a reactor coolant pressure change, to the corresponding effect on neutron production. This coefficient is of much less significance than the moderator temperature coefficient. A change of 50 psi in pressure has approximately the same effect on reactivity as a one half degree change in moderator temperature. This coefficient can be determined from the moderator temperature coefficient by relating change in pressure to the corresponding change in density. The typical moderator pressure coefficient may be negative over a portion of the moderator temperature range at beginning of life (BOL) (-0.004 pcm/psi) but is always positive at operating conditions and becomes more positive during life (+0.3 pcm/psi, at end of life).

4.3.2.3.2.3 Moderator Void Coefficient

The moderator void coefficient relates the change in neutron multiplication to the presence of voids in the moderator. In a PWR, this coefficient is not very significant because of the low void content in the coolant. The core void content is less than one-half of one percent and is due to local or statistical boiling. The typical void coefficient varies from 50 pcm/percent void at BOL and at low temperatures to minus 250 pcm/percent void at EOL and at operating temperatures. The void coefficient at operating temperature becomes more negative with fuel burnup.

4.3.2.3.3 Power Coefficient

The combined effect of moderator temperature and fuel temperature change as the core power level changes is called the total power coefficient and is expressed in terms of reactivity change per percent power change. Since a three-dimensional calculation is performed in determining total power coefficients and total power defects, the axial redistribution reactivity component described in subsection 4.3.2.4.3 is implicitly included. A typical power coefficient at beginning of life (BOL) and end of life (EOL) conditions is given in Figure 4.3-25.

The total power coefficient becomes more negative with burnup, reflecting the combined effect of moderator and fuel temperature coefficients with burnup. The power defect (integral reactivity effect) at BOL and EOL is given in Figure 4.3-26.

4.3.2.3.4 Comparison of Calculated and Experimental Reactivity Coefficients

Subsection 4.3.3 describes the comparison of calculated and experimental reactivity coefficients in detail.

Experimental evaluation of the reactivity coefficients will be performed 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 typical best-estimate values for the reactivity coefficients for the initial cycle. 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 non-uniformity 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 4.3-18 through 4.3-26 are typical best-estimate values calculated for the initial cycle. Limiting values are chosen to encompass the best-estimate reactivity coefficients, including the uncertainties given in subsection 4.3.3.3 over appropriate operating conditions. 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 for the initial cycle is shown in Table 4.3-2. 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 assumed in the analysis. The need for a reevaluation of any accident in a subsequent cycle is contingent upon whether 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 subsection 4.3.3.3. Control rod requirements are given in Table 4.3-3 for the initial cycle and for a hypothetical equilibrium cycle, since these are markedly different. These latter numbers are provided for information only.

4.3.2.4 Control Requirements

To establish the required shutdown margin stated in the COLR 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 the initial cycle. For 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 shutdown condition. The minimum required shutdown margin is given in the COLR.

The ability to meet the shutdown margin requirements for hot conditions is demonstrated for the initial cycle and for an equilibrium reload cycle by performing a bounding calculation, the

results of which are shown in Table 4.3-3. Table 4.3-3 compares 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 seven percent for analytic uncertainties which assumes the use of silver-indium-cadmium rod cluster control assemblies. Use of a seven percent uncertainty allowance on rod cluster control assembly worth is discussed and shown to be acceptable in WCAP-9217 (Reference 17). 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.

Any available negative reactivity insertion from withdrawn tungsten GRCAs is conservatively excluded when determining the available shutdown margin at hot operating conditions, even though all GRCAs are released into the core on a reactor trip (Reference 70). Only silver-indium-cadmium control rods are assumed to insert when the reactor is tripped for purposes of demonstrating that adequate shutdown margin is available at hot operating conditions. As such, the use of a seven percent uncertainty allowance for the credited trip rod worth remains appropriate in Table 4.3-3. After the reactor is brought to a shut down condition, the presence of GRCAs which are confirmed to be inserted and which have met the applicable physics testing acceptance criteria may be credited in confirming that the required shutdown margin is maintained during any cooldown period and as the result of long term xenon decay (Reference 70).

During plant operation, the available shutdown margin for hot operating conditions is continuously confirmed by the online monitoring system, by comparing the operating soluble boron concentration at current core conditions to the shutdown boron concentration that would be required immediately following a reactor trip from those conditions. The required shutdown boron concentration used in this type of calculation is conservatively determined at the target shutdown reactivity condition, assuming that all control rods insert except for the 16 tungsten GRCAs and the highest worth silver-indium-cadmium RCCA.

The control rods are required to provide sufficient reactivity to account for the power defect from full power to zero power and to provide the required shutdown margin. The reactivity addition resulting from power reduction consists of contributions from Doppler effect, moderator temperature, flux redistribution, and reduction in void content as discussed below.

4.3.2.4.1 Doppler Effect

The Doppler effect arises from the broadening of U-238 and Pu-240 resonance cross-sections with an increase in effective pellet temperature. This effect is most noticeable over the range of zero power to full power due to the large pellet temperature increase with power generation. The Doppler effect is implicitly included in the total power defects shown in Table 4.3-3.

4.3.2.4.2 Variable Average Moderator Temperature

When the core is shut down to the hot zero-power condition, the average moderator temperature changes from the equilibrium full-load value determined by the steam generator and turbine characteristics (such as steam pressure, heat transfer, tube fouling) to the

equilibrium no-load value, which is based on the steam generator shell side design pressure. The design change in temperature is conservatively increased to account for the control system dead band and measurement errors.

When the moderator coefficient is negative, there is a reactivity addition with power reduction. The moderator coefficient becomes more negative as the fuel depletes because the boron concentration is reduced. This effect is the major contributor to the increased requirement at EOL. The change in average moderator temperature is implicitly included in the total power defects shown in Table 4.3-3.

4.3.2.4.3 Redistribution

During full-power operation, the coolant density decreases with core height. This, together with partial insertion of control rods, results in less fuel depletion near the top of the core. Under steady-state conditions, the relative power distribution will be slightly asymmetric toward the bottom of the core. On the other hand, at hot zero-power conditions, the coolant density is uniform up the core, and there is no flattening due to Doppler effect. The result will be a flux distribution which at zero power can be skewed toward the top of the core. Since a three-dimensional calculation is performed in determining total power defect, flux redistribution is implicitly included in this calculation. The three-dimensional total power defects specified in Table 4.3-3 were calculated including the use of a conservatively skewed adverse axial xenon distribution which increases the redistribution effect.

4.3.2.4.4 Void Content

A small void content in the core is due to nucleate boiling at full power. The void collapse coincident with power reduction makes a small positive reactivity contribution which has been added to the calculated total power defects shown in Table 4.3-3.

4.3.2.4.5 Rod Insertion Allowance

The MSHIM and AO banks are operated within a prescribed band of travel to compensate for changes in temperature and axial offset which are caused by fuel depletion and power maneuvers as described in Section 4.3.2.4.16. In calculating the available shutdown margin at hot operating conditions, the pre-trip control rod insertion can affect both the available trip rod worth and the total power defect control requirements. In addition, since the tungsten GRCAs are assumed not to insert on a reactor trip (Reference 70), the initial gray rod positions assumed prior to the trip can also have a small effect on the worth of the silver-indium-cadmium control rods that insert after the trip. In the bounding calculations shown in Table 4.3-3, the effect of the most limiting allowed control rod insertion is implicitly included in the calculated trip rod worth and total power defect values reported in the table. The most limiting allowed control rod insertion was determined by performing a series of three-dimensional shutdown margin calculations over the range of allowed control rod motion, and selecting the conditions which resulted in the minimum calculated shutdown margin.

4.3.2.4.6 Installed Excess Reactivity for Depletion

Excess reactivity is installed at the beginning of each cycle to provide sufficient reactivity to compensate for fuel depletion and fission product buildup throughout the cycle. This reactivity is controlled by the addition of soluble boron to the coolant, control rod insertion, and by burnable absorbers when necessary. The soluble boron concentration for several core configurations and the unit boron worth are given in Tables 4.3-1 and 4.3-2 for the initial cycle. Since the excess reactivity for burnup is balanced during operation by negative reactivity from the above sources, it is not included in control rod requirements.

4.3.2.4.7 Xenon and Samarium Poisoning

Changes in xenon and samarium concentrations in the core occur at a sufficiently slow rate, even following rapid power level changes, that the resulting reactivity change can be controlled by changing the gray and/or control rod insertion. (Also see subsection 4.3.2.4.16).

4.3.2.4.8 pH Effects

Changes in reactivity due to a change in coolant pH, if any, are sufficiently small in magnitude and occur slowly enough to be controlled by the boron system WCAP-3696-8 (Reference 18).

4.3.2.4.9 Experimental Confirmation

Following a normal shutdown, the total core reactivity change during cooldown with a stuck rod has been measured on a 121-assembly, 10-foot-high core and a 121-assembly, 12-foot-high core. In each case, the core was allowed to cool down until it reached criticality simulating the steam line break accident. For the 10-foot core, the total reactivity change associated with the cooldown is over predicted by about 0.3-percent $\Delta\rho$ with respect to the measured result. This represents an error of about five percent in the total reactivity change and is about half the uncertainty allowance for this quantity. For the 12-foot core, the difference between the measured and predicted reactivity change is an even smaller 0.2 percent $\Delta\rho$. These measurements and others demonstrate the capability of the methods described in subsection 4.3.3.

4.3.2.4.10 Control

Core reactivity is controlled by means of a chemical poison dissolved in the coolant, rod cluster control assemblies, gray rod cluster assemblies and burnable absorbers as described below.

4.3.2.4.11 Chemical Shim

Boron in solution as boric acid is used to control relatively slow reactivity changes associated with:

- The moderator temperature defect in going from cold shutdown at ambient temperature to the hot operating temperature at zero power

- Transient xenon and samarium reactivity effects, following power changes
- The reactivity effects of fissile inventory depletion and buildup of long-life fission products
- The depletion of the burnable absorbers

The boron concentrations for various core conditions are presented in Table 4.3-2 for the initial cycle.

4.3.2.4.12 Rod Cluster Control Assemblies

The number of rod cluster control assemblies is shown in Table 4.3-1. The rod cluster control assemblies are used for shutdown and control purposes to offset fast reactivity changes associated with:

- The required shutdown margin in the hot zero power, stuck rod condition
- The reactivity compensation as a result of an increase in power above hot zero power (power defect, including Doppler and moderator reactivity changes)
- Unprogrammed fluctuations in boron concentration, coolant temperature, or xenon concentration (with rods not exceeding the allowable rod insertion limits)
- Reactivity changes resulting from load changes

The allowed 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 an alarm 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 the analyses is used in determining power distribution factors and in determining the maximum worth of an inserted rod cluster control assembly ejection accident. For further discussion, refer to the technical specifications on 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 4.3-27. Shutdown rod cluster control assemblies are withdrawn before withdrawal of the control and AO banks is initiated. The approach to critical is initiated by using the chemical and volume control system to establish an appropriate boron concentration based upon the estimated critical condition then withdrawing the AO bank above the zero power insertion limit and finally withdrawing the control banks sequentially. The limits of rod insertion and further discussion on the basis for rod insertion limits are provided in the COLR.

4.3.2.4.13 Gray Rod Cluster Assemblies

The rod cluster control assembly control banks include four gray rod banks consisting of gray rod cluster assemblies (GRCAs). Gray rod cluster assemblies consist of 24 rodlets fastened at the top end to a common hub or spider. Geometrically, it is the same as a rod cluster control assembly except that the GRCA design uses tungsten encapsulated in a nickel-chromium-iron Alloy 718 sleeve as an absorber. The term gray rod refers to the reduced reactivity worth relative to that of a rod cluster control assembly consisting of 24 silver-indium-cadmium rodlets. The gray rod cluster assemblies are used in base load operation and load follow maneuvering and provide a mechanical shim reactivity mechanism which reduces the need for changes to the concentration of soluble boron (that is, chemical shim).

4.3.2.4.14 Burnable Absorbers

Discrete burnable absorber rods and integral fuel burnable absorber rods will be used to provide partial control of the excess reactivity available during the first operating cycle. In doing so, the burnable absorber loading controls peaking factors and prevents the moderator temperature coefficient from being positive at normal operating conditions. The burnable absorbers perform this function by reducing the requirement for soluble boron in the moderator at the beginning of the fuel cycle, as described previously. For purposes of illustration, the initial cycle burnable absorber pattern is shown in Figure 4.3-5. Figures 4.3-4a and 4.3-4b show the burnable absorber distribution within a fuel assembly for several burnable absorber patterns used in the 17 x 17 array. The boron in the rods is depleted with burnup but at a slow rate so that the peaking factor limits are not exceeded and the resulting critical concentration of soluble boron is such that the moderator temperature coefficient remains within the limits stated above for power operating conditions.

4.3.2.4.15 Peak Xenon Startup

Compensation for the peak xenon buildup may be 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 can 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 primarily accomplished using control rod motion alone, as required. Control rod motion is limited by the control rod insertion limits as provided in the COLR and discussed in subsections 4.3.2.4.12 and 4.3.2.4.5. The power distribution is maintained within acceptable limits through limitations on control rod insertion. Reactivity changes due to the changing xenon concentration are also controlled by rod motion. The soluble boron concentration may also be changed during large load change maneuvers or during extended reduced power operation to maintain the control rods in a more optimum range for power distribution control.

Rapid power increases (five percent/min) from part power during load follow operation are accomplished with rod motion.

The rod control system is designed to automatically provide the power and temperature control described above 30 percent rated power for most of the cycle length without the need to change boron concentration as a result of the load maneuver. The automated mode of operation is referred to as mechanical shim (MSHIM) because of the usage of mechanical means to control reactivity and power distribution simultaneously. MSHIM operation allows load maneuvering without boron change because of the degree of allowed insertion of the control banks in conjunction with the independent power distribution control of the axial offset (AO) control bank. The worth and overlap of the MA, MB, MC, MD, M1, and M2 control banks are designed such that the AO control bank insertion will always result in a monotonically decreasing axial offset. MSHIM operation uses the MA, MB, MC, MD, M1, and M2 control banks to maintain the programmed coolant average temperature throughout the operating power range. The AO control bank is independently modulated by the rod control system to maintain a nearly constant axial offset throughout the operating power range. The target axial offset used during MSHIM operation is established at a more negative value than the axial offset associated with the all rods out condition. The negative bias is necessary to maintain both positive and negative axial offset control effectiveness by the AO-bank. Operation with gray control rod banks (MA, MB, MC, and MD) inserted has less of an effect on the core axial power distribution than insertion of the black control rods banks (M1 and M2) and results in a smaller negative bias in the target axial offset. Load change operations that are large enough to require a black control rod bank to enter the core may require a more negative target axial offset to accomplish. However, the boron system can optionally be used to maintain operation in the more optimum range of gray rod motion during such maneuvers. The degree of control rod insertion under MSHIM operation allows rapid return to power without the need to change boron concentration.

Extended base load operation is performed by controlling axial offset to the target value using the AO control bank, and by controlling the coolant average temperature to the programmed value with the M-banks. Boron concentration changes are made periodically as the fuel depletes to reposition the M-banks and allow for a periodic exchange of the gray rod bank insertion sequence. MSHIM load follow and base load operations (including the gray rod bank insertion sequence exchanges) are considered Condition I normal operations.

4.3.2.4.17 Burnup

Control of the excess reactivity for burnup is accomplished using soluble boron, control rod insertion, and/or burnable absorbers. The boron concentration is limited during operating conditions to maintain the moderator temperature coefficient within its specified limits. A sufficient burnable absorber loading is installed at the beginning of a cycle to give the desired cycle lifetime, without exceeding the boron concentration limit. The end of a fuel cycle is reached when the soluble boron concentration approaches the practical minimum boron concentration in the range of 0 to 10 ppm.

4.3.2.4.18 Rapid Power Reduction System

The reactor power control system is designed with the capability of responding to full load rejection without initiating a reactor trip using the normal rod control system, reactor control system, and the rapid power reduction system. Load rejections requiring greater than a fifty percent reduction of rated thermal power initiate the rapid power reduction system. The

rapid power reduction system utilizes preselected control rod groups and/or banks which are intentionally tripped to rapidly reduce reactor power into a range where the rod control and reactor control systems are sufficient to maintain stable plant operation. The consequences of accidental or inappropriate actuation of the rapid power reduction system is included in the cycle specific safety analysis and licensing process.

4.3.2.5 Control Rod Patterns and Reactivity Worth

The rod cluster control assemblies are designated by function as the control groups and the shutdown groups. The terms group and bank are used synonymously to describe a particular grouping of control assemblies. The rod cluster control assembly patterns are displayed in Figure 4.3-27. The control banks are labeled MA, MB, MC, MD, M1, M2, and AO with the MA, MB, MC, and MD banks comprised of gray rod cluster assemblies; and the shutdown banks are labeled S1, S2, S3, and S4. Each bank of more than four rod cluster control assemblies, although operated and controlled as a unit, is composed of two or more subgroups. The axial position of the rod cluster control assemblies may be controlled manually or automatically. The rod cluster control assemblies are dropped into the core following actuation of reactor trip signals.

Two criteria have been employed for selection of the control groups. First, the total reactivity worth must be adequate to meet the requirements specified in Table 4.3-3. Second, in view of the fact that these rods may be partially inserted at power operation, the total power peaking factor should be low enough to meet the power capability requirements. Analyses indicate that the first requirement can be met either by a single group or by two or more banks whose total worth equals at least the required amount. The axial power shape is more peaked following movement of a single group of rods worth three to four percent $\Delta\rho$. Therefore, control bank rod cluster control assemblies have been separated into several bank groupings. Typical control bank worth for the initial cycle are shown in Table 4.3-2.

The position of control banks for criticality under any reactor condition is determined by the concentration of boron in the coolant. On an approach to criticality, boron is adjusted so that criticality will be achieved with control rods above the insertion limit set by shutdown and other considerations. (See the technical specifications and COLR). Early in the cycle, there may also be a withdrawal limit at low power to maintain the moderator temperature coefficient within the specified limits for that power level.

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

Allowable deviations due to misaligned control rods are discussed in the technical specifications.

A representative differential rod worth calculation for two banks of control rods withdrawn simultaneously (rod withdrawal accident) is given in Figure 4.3-28.

Calculation of control rod reactivity worth versus time following reactor trip involves both control rod velocity and differential reactivity worth. A representative example of the rod position versus time of travel after rod release is given in Figure 4.3-29. The actual rod position versus time of travel used in the safety analysis is given in Section 15.0. For nuclear

design purposes, the reactivity worth versus rod position is calculated by a series of steady-state calculations at various control positions, assuming the 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. A representative result of these calculations is shown in Figure 4.3-30.

The shutdown groups provide additional negative reactivity to establish adequate shutdown margin. Shutdown margin is the amount by which the core would be subcritical at hot shutdown if the rod cluster control assemblies were tripped, but assuming that the highest worth assembly remained fully withdrawn and no changes in xenon or boron took place. The loss of control rod worth due to the depletion of the absorber material is negligible.

The values given in Table 4.3-3 show that the available reactivity in withdrawn rod cluster control assemblies provides the design bases minimum shutdown margin, allowing for the highest worth cluster to be at its fully withdrawn position. An allowance for the uncertainty in the calculated worth of N-1 rods is made before determination of the shutdown margin.

4.3.2.6 Criticality of the Reactor During Refueling

The basis for maintaining the reactor subcritical during refueling is presented in subsection 4.3.1.5, and a discussion of how control requirements are met is given in subsections 4.3.2.4 and 4.3.2.5.

4.3.2.6.1 Criticality Design Method Outside the Reactor

Criticality of fuel assemblies outside the reactor is precluded by adequate design of fuel transfer, shipping, and storage facilities and by administrative control procedures. The two principal methods of preventing criticality are limiting the fuel assembly array size and limiting assembly interaction by fixing the minimum separation between assemblies and/or inserting neutron poisons between assemblies. The details of the methodology used for the new fuel rack and spent fuel rack criticality analysis are included in the Chapter 9.1 references.

The design criteria are consistent with General Design Criterion (GDC) 62, Reference 19, and NRC guidance given in Reference 20. The applicable 10 CFR Part 50.68 requirements are as follows:

1. The maximum K-effective value, including all biases and uncertainties, must be less than 0.95 with soluble boron credit and less than 1.0 with full density unborated water. Note this design criterion is provided in 10 CFR Part 50.68, Item 4 of Paragraph b. Note that the specific terminology is:

   "If no credit for soluble boron is taken, the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with unborated water. If credit is taken for soluble boron, the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent

probability, 95 percent confidence level, if flooded with borated water, and the k-effective must remain below 1.0 (subcritical), at a 95 percent probability, 95 percent confidence level, if flooded with unborated water."

2. The maximum enrichment of fresh fuel assemblies must be less than or equal to 5.0 weight-percent U-235. Note this design criterion is provided in 10 CFR Part 50.68, Item 7 of Paragraph b. Note that the specific terminology is:

   "The maximum nominal U–235 enrichment of the fresh fuel assemblies is limited to five (5.0) percent by weight."

The following conditions are assumed in meeting this design bases:

- The fuel assembly contains the highest enrichment authorized without any control rods or non-integral burnable absorber(s) and is at its most reactive point in life.
- For flooded conditions, the moderator is pure water at the temperature within the design limits which yields the largest reactivity.
- The array is either infinite in lateral extent or is surrounded by a conservatively chosen reflector, whichever is appropriate for the design.
- Mechanical uncertainties are treated by combining both the worst-case bounding value and sensitivity study approaches.
- Credit is taken for the neutron absorption in structural materials and in solid materials added specifically for neutron absorption.

Fuel depletion analyses during core operation were performed with CASMO-4 (using the 70-group cross-section library), a two-dimensional multigroup transport theory code based on capture probabilities (Reference 53). CASMO-4 is used to determine the isotopic composition of the spent fuel. In addition, the CASMO-4 calculations are restarted in the storage rack geometry, yielding the two-dimensional infinite multiplication factor (k_{inf}) for the storage rack to determine the reactivity effect of fuel and rack tolerances, temperature variation, and to perform various studies.

The design method which determines the criticality safety of fuel assemblies outside the reactor uses the MCNP4a code (Reference 21), with continuous energy cross-sections based on ENDF/B-V and ENDF/B-VI.

A set of 62 critical experiments has been analyzed using the above method to demonstrate its applicability to criticality analysis and to establish the method bias and uncertainty. The benchmark experiments cover a wide range of geometries, materials, and enrichments, all of them adequate for qualifying methods to analyze light water reactor lattices (References 22 to 28, and 65 to 68).

The analysis of the 62 critical experiments results in an average K_{eff} of 0.9991. Comparison with the measured values results in a method bias of 0.0009. The standard deviation of the set of reactivities is 0.0011. The 95/95 tolerance factor is conservatively set to 2.0.

The analytical methods employed herein conform with ANSI N18.2 (Reference 3), Section 5.7, Fuel Handling System; ANSI N16.9 (Reference 29), NRC Standard Review Plan, subsection 9.1.2, the NRC guidance, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications" (Reference 30).

4.3.2.6.2 Soluble Boron Credit Methodology

The minimum soluble boron requirement under normal and accident conditions must be determined to show that the reactivity of the spent fuel racks remains below 0.95. This is achieved by crediting a discrete amount of soluble boron and then determining by linear interpolation the appropriate amount of soluble boron necessary to reduce the maximum Keff to 0.95 with all uncertainties and biases included..

4.3.2.7 Stability

4.3.2.7.1 Introduction

The stability of the PWR cores against xenon-induced spatial oscillations and the control of such transients are discussed extensively in References 11, 31, 32, and 33. A summary of these reports is given in the following discussion, and the design bases are given in subsection 4.3.1.6.

In a large reactor core, xenon-induced oscillations can take place with no corresponding change in the total power of the core. The oscillation may be caused by a power shift in the core which occurs rapidly by comparison with the xenon-iodine time constants. Such a power shift occurs in the axial direction when a plant load change is made by control rod motion and results in a change in the moderator density and fuel temperature distributions. Such a power shift could occur in the diametral plane of the core as a result of abnormal control action.

Due to the negative power coefficient of reactivity, PWR cores are inherently stable to oscillations in total power. Protection against total power instabilities is provided by the control and protection system, as described in Section 7.7. Hence, the discussion on the core stability will be limited to xenon-induced spatial oscillations.

4.3.2.7.2 Stability Index

Power distributions, either in the axial direction or in the X-Y plane, can undergo oscillations due to perturbations introduced in the equilibrium distributions without changing the total core power. The harmonics and the stability of the core against xenon-induced oscillations can be determined in terms of the eigenvalue of the first flux harmonics. Writing the eigenvalue ξ of the first flux harmonic as:

$$\xi = b + ic \tag{1}$$

Then b is defined as the stability index and $T = 2\pi / c$ as the oscillation period of the first harmonic. The time dependence of the first harmonic $\delta \phi$ in the power distribution can now be represented as:

$$\delta\phi(t) = A e^{\xi t} = a e^{bt} \cos ct \qquad (2)$$

where A and a are constants. The stability index can also be obtained approximately by:

$$b = \frac{1}{T} \ln \frac{A_{n+1}}{A_n} \qquad (3)$$

where A_n and A_{n+1} are the successive peak amplitudes of the oscillation and T is the time period between the successive peaks.

4.3.2.7.3 Prediction of the Core Stability

The core described in this report has an active fuel length that is 24 inches longer (nominal) than that for previous Westinghouse PWRs licensed in the U.S. with 157 fuel assemblies. For this reason, it is expected that this core will be as stable as the 12-foot designs with respect to radial and diametral xenon oscillations since the radial core dimensions have not changed. This core will be slightly less stable than the 12-foot, 157 assembly cores with respect to axial xenon oscillations because the active core height has been increased by 24 inches. The effect of this increase will be to decrease the burnup at which the axial stability index becomes zero (Section 4.3.2.7.4 below). The moderator temperature coefficients and the Doppler temperature coefficients of reactivity will be similar to those of previous designs. Control banks included in the core design are sufficient to dampen any xenon oscillations that may occur. Free axial xenon oscillations are not allowed to occur for a core of any height, except during special tests as described in Section 4.3.2.7.4.

4.3.2.7.4 Stability Measurements

4.3.2.7.4.1 Axial Measurements

Two axial xenon transient tests conducted in a PWR with a core height of 12 feet and 121 fuel assemblies are reported in WCAP-7964 (Reference 34) and are discussed here. The tests were performed at approximately 10 percent and 50 percent of cycle life.

Both a free-running oscillation test and a controlled test were performed during the first test. The second test at mid-cycle consisted of a free-running oscillation test only. In each of the free-running oscillation tests, a perturbation was introduced to the equilibrium power distribution through an impulse motion of the lead control bank and the subsequent oscillation period was monitored. In the controlled test conducted early in the cycle, the part-length rods were used to follow the oscillations to maintain an axial offset within the prescribed limits. The axial offset of power was obtained from the ex-core ion chamber readings (which had been calibrated against the in-core flux maps) as a function of time for both free-running tests, as shown in Figure 12 of WCAP-7964 (Reference 34)

The total core power was maintained constant during these spatial xenon tests, and the stability index and the oscillation period were obtained from a least-square fit of the axial offset data in the form of equation 2. The axial offset of power is the quantity that properly represents the axial
stability in the sense that it essentially eliminates any contribution from even-order harmonics, including the fundamental mode. The conclusions of the tests follow:

- The core was stable against induced axial xenon transients, at the core average burnups of both 1550 MWD/MTU and 7700 MWD/MTU. The measured stability indices are -0.041 h^{-1} for the first test and - 0.014 h^{-1} for the second test. The corresponding oscillation periods are 32.4 and 27.2 hours, respectively.

- The reactor core becomes less stable as fuel burnup progresses, and the axial stability index is essentially zero at 12,000 MWD/MTU. However, the movable control rod systems can control axial oscillations, as described in subsection 4.3.2.7.3.

4.3.2.7.4.2 Measurements in the X-Y Plane

Two X-Y xenon oscillation tests were performed at a PWR plant with a core height of 12 feet and 157 fuel assemblies. The first test was conducted at a core average burnup of 1540 MWD/MTU and the second at a core average burnup of 12,900 MWD/MTU. Both of the X-Y xenon tests show that the core was stable in the X-Y plane at both burnups. The second test shows that the core became more stable as the fuel burnup increased, and Westinghouse PWRs with 121 and 157 assemblies are stable throughout their burnup cycles. The results of these tests are applicable to the 157-assembly AP1000 core, as discussed in subsection 4.3.2.7.3.

In each of the two X-Y tests, a perturbation was introduced to the equilibrium power distribution through an impulse motion of one rod cluster control unit located along the diagonal axis. Following the perturbation, the uncontrolled oscillation was monitored, using the movable detector and thermocouple system and the ex-core power range detectors. The quadrant tilt difference (QTD) is the quantity that properly represents the diametral oscillation in the X-Y plane of the reactor core in that the differences of the quadrant average powers over two symmetrically opposite quadrants essentially eliminates the contribution to the oscillation from the azimuthal mode. The quadrant tilt difference data were fitted in the form of equation 2 of subsection 4.3.2.7.2 through a least-square method. A stability index of - 0.076 hr^{-1} (per hour) with a period of 29.6 hr was obtained from the thermocouple data shown in Figure 4.3-31.

It was observed in the second X-Y xenon test that the PWR core with 157 fuel assemblies had become more stable due to an increased fuel depletion, and the stability index was not determined.

4.3.2.7.5 Comparison of Calculations with Measurements

The direct simulation of axial offset data was carried out using a licensed one-dimensional code (WCAP-7048-P-A (Reference 35)). The analysis of the X-Y xenon transient tests was performed in an X-Y geometry, using a licensed few group two-dimensional code

(WCAP-7213-A (Reference 36)). Both of these codes solve the two-group, time-dependent neutron diffusion equation with time-dependent xenon and iodine concentrations. The fuel temperature and moderator density feedback is limited to a steady-state model. The X-Y calculations were performed in an average enthalpy plane.

The detailed experimental data during the tests, including the reactor power level, the enthalpy rise, and the impulse motion of the control rod assembly, as well as the plant follow burnup data, were closely simulated in the study.

The results of the stability calculation for the axial tests are compared with the experimental data in Table 4.3-5. The calculations show conservative results for both of the axial tests with a margin of approximately 0.01 hr^{-1} in the stability index.

An analytical simulation of the first X-Y xenon oscillation test shows a calculated stability index of - 0.081 hr^{-1}, in good agreement with the measured value of - 0.076 hr^{-1}. As indicated earlier, the second X-Y xenon test showed that the core had become more stable compared to the first test, and no evaluation of the stability index was attempted. This increase in the core stability in the X-Y plane due to increased fuel burnup is due mainly to the increased magnitude of the negative moderator temperature coefficient.

Previous studies of the physics of xenon oscillations, including three-dimensional analysis, are reported in a series of topical reports (References 31, 32, and 33). A more detailed description of the experimental results and analysis of the axial and X-Y xenon transient tests is presented in WCAP-7964 (Reference 34) and Section 1 of WCAP-8768 (Reference 37).

4.3.2.7.6 Stability Control and Protection

The online monitoring system provides continuous indication of current power distributions and provides guidance to the plant operator as to the timing and most appropriate action(s) to maintain stable axial power distributions. In the event the online monitoring system is out of service, the ex-core detector system is utilized to provide indications of xenon-induced spatial oscillations. The readings from the ex-core detectors are available to the operator and also form part of the protection system.

4.3.2.7.6.1 Axial Power Distribution

The rod control system automatically maintains axial power distribution within very tight axial offset bands as part of normal operation. The AO control bank is specifically designed with sufficient worth to be capable of maintaining essentially constant axial offset over the power operating range. The rod control system is also allowed to be operated in manual control in which case the operator is instructed to maintain an axial offset within a prescribed operating band, based on the ex-core detector readings. Should the axial offset be permitted to move far enough outside this band, the protection limit is encroached, and the turbine power is automatically reduced or a reactor trip signal generated, or both.

As fuel burnup progresses, PWR cores become less stable to axial xenon oscillations. However, free xenon oscillations are not allowed to occur, except for special tests. The AO control bank is sufficient to dampen and control any axial xenon oscillations present. Should

the axial offset be inadvertently permitted to move far enough outside the allowed band due to an axial xenon oscillation or for any other reason, the OTΔT and/or OPΔT protection setpoint including the axial offset compensation is reached and the turbine power is automatically reduced and/or a reactor trip signal is generated.

4.3.2.7.6.2 Radial Power Distribution

The core described herein is calculated to be stable against X-Y xenon-induced oscillations during the core life.

The X-Y stability of large PWRs has been further verified as part of the startup physics test program for PWR cores with 193 fuel assemblies. The measured X-Y stability of the cores with 157 and 193 assemblies was in close agreement with the calculated stability, as discussed in subsections 4.3.2.7.4 and 4.3.2.7.5. In the unlikely event that X-Y oscillations occur, backup actions are possible and would be implemented, if necessary, to increase the natural stability of the core. This is based on the fact that several actions could be taken to make the moderator temperature coefficient more negative, which would increase the stability of the core in the X-Y plane.

Provisions for protection against non-symmetric perturbations in the X-Y power distribution that could result from equipment malfunctions are made in the protection system design. This includes control rod drop, rod misalignment, and asymmetric loss of coolant flow.

A more detailed discussion of the power distribution control in PWR cores is presented in WCAP-7811 (Reference 11) and WCAP-8385 (Reference 12).

4.3.2.8 Vessel Irradiation

A review of the methods and analyses used in the determination of neutron and gamma ray flux attenuation between the core and the pressure vessel is provided below. A more complete discussion on the pressure vessel irradiation and surveillance program is given in Section 5.3.

The materials that serve to attenuate neutrons originating in the core and gamma rays from both the core and structural components consist of the core shroud, core barrel and associated water annuli. These are within the region between the core and the pressure vessel.

In general, few group neutron diffusion theory codes are used to determine fission power density distributions within the active core, and the accuracy of these analyses is verified by in-core measurements on operating reactors. Region and rodwise power-sharing information from the core calculations is then used as source information in two-dimensional transport calculations which compute the flux distributions throughout the reactor.

The neutron flux distribution and spectrum in the various structural components vary significantly from the core to the pressure vessel. Representative values of the neutron flux distribution and spectrum are presented in Table 4.3-6.

As discussed in Section 5.3, the irradiation surveillance program utilizes actual test samples to verify the accuracy of the calculated fluxes at the vessel.

4.3.3 Analytical Methods

Calculations required in nuclear design consist of three distinct types, which are performed in sequence:

1. Determination of effective fuel temperatures
2. Generation of few-group cross sections
3. Space-dependent, few-group diffusion calculations

These calculations are carried out by computer codes which can be executed individually. Most of the codes required have been linked to form an automated design sequence which minimizes design time, avoids errors in transcription of data, and standardizes the design methods.

4.3.3.1 Fuel Temperature (Doppler) Calculations

Temperatures vary radially within the fuel rod, depending on the heat generation rate in the pellet; the conductivity of the materials in the pellet, gap, and clad; and the temperature of the coolant.

The fuel temperatures for use in most nuclear design Doppler calculations are obtained from a simplified version of the Westinghouse fuel rod design model described in subsection 4.2.1.3, which considers the effect of radial variation of pellet conductivity, expansion coefficient and heat generation rate, elastic deflection of the clad, and a gap conductance which depends on the initial fill gas, the hot open gap dimension, and the fraction of the pellet over which the gap is closed. The fraction of the gap assumed closed represents an empirical adjustment used to produce close agreement with observed reactivity data at beginning of life. Further gap closure occurs with burnup and accounts for the decrease in Doppler defect with burnup which has been observed in operating plants. For detailed calculations of the Doppler coefficient, such as for use in xenon stability calculations, a more sophisticated temperature model is used, which accounts for the effects of fuel swelling, fission gas release, and plastic clad deformation.

Radial power distributions in the pellet as a function of burnup are obtained from LASER (WCAP-6073, Reference 38) calculations.

The effective U-238 temperature for resonance absorption is obtained from the radial temperature distribution by applying a radially dependent weighing function. The weighing function was determined from REPAD (WCAP-2048, Reference 39) Monte Carlo calculations of resonance escape probabilities in several steady-state and transient temperature distributions. In each case, a flat pellet temperature was determined which produced the same resonance escape probability as the actual distribution. The weighing function was empirically determined from these results.

The effective Pu-240 temperature for resonance absorption is determined by a convolution of the radial distribution of Pu-240 densities from LASER burnup calculations and the radial weighing function. The resulting temperature is burnup dependent, but the difference between U-238 and Pu-240 temperatures, in terms of reactivity effects, is small.

The effective pellet temperature for pellet dimensional change is that value which produces the same outer pellet radius in a virgin pellet as that obtained from the temperature model. The effective clad temperature for dimensional change is its average value.

The temperature calculational model has been validated by plant Doppler defect data, as shown in Table 4.3-7, and Doppler coefficient data, as shown in Figure 4.3-32. Stability index measurements also provide a sensitive measure of the Doppler coefficient near full power (subsection 4.3.2.7).

4.3.3.2 Macroscopic Group Constants

PHOENIX-P (WCAP-11596-P-A, Reference 40) and PARAGON (WCAP-16045-P-A, Reference 69) have been used for generating the macroscopic cross sections needed for the spatial few group codes. PHOENIX-P, PARAGON, or other NRC approved lattice codes will be used for reload designs.

PHOENIX-P has been approved by the NRC as a lattice code for the generation of macroscopic and microscopic few group cross sections for PWR analysis. (See WCAP-11596-P-A, Reference 40). PHOENIX-P is a two-dimensional, multigroup, transport-based lattice code capable of providing necessary data for PWR analysis. Since it is a dimensional lattice code, PHOENIX-P does not rely on pre-determined spatial/spectral interaction assumptions for the heterogeneous fuel lattice and can provide a more accurate multigroup spatial flux solution than versions (ARK) of LEOPARD/CINDER.

The solution for the detailed spatial flux and energy distribution is divided into two major steps in PHOENIX-P (See References 40 and 41). First, a two-dimensional fine energy group nodal solution is obtained, coupling individual subcell regions (e.g., pellet, clad and moderator) as well as surrounding pins, using a method based on Carlvik's collision probability approach and heterogeneous response fluxes which preserve the heterogeneous nature of the pin cells and their surroundings. The nodal solution provides an accurate and detailed local flux distribution, which is then used to homogenize the pin cells spatially to few groups.

Then, a standard S_4 discrete ordinates calculation solves for the angular distribution, based on the group-collapsed and homogenized cross sections from the first step. These S_4 fluxes normalize the detailed spatial and energy nodal fluxes, which are then used to compute reaction rates, power distributions and to deplete the fuel and burnable absorbers. A standard B1 calculation evaluates the fundamental mode critical spectrum, providing an improved fast diffusion coefficient for the core spatial codes.

PHOENIX-P employs a 70 energy group library derived mainly from the ENDF/B-VI files (Reference 71). This library was designed to capture the integral properties of the multigroup data properly during group collapse and to model important resonance parameters properly. It contains neutronics data necessary for modelling fuel, fission products, cladding and structural materials, coolant, and control and burnable absorber materials present in PWRs.

Group constants for burnable absorber cells, RCCA cells, guide thimbles and instrumentation thimbles, or other non-fuel cells, can be obtained directly from PHOENIX-P without any adjustments such as those required in the cell or 1D lattice codes.

PHOENIX-P has been validated through an extensive qualification effort which includes calculation-measurement comparison of the Strawbridge-Barry critical experiments (See References 42 and 43), the KRITZ high temperature criticals (Reference 44), the AEC sponsored B&W criticals (References 45 through 47) and measured actinide isotopic data from fuel pins irradiated in the Saxton and Yankee Rowe cores (References 48 through 52). In addition, calculation-measurement comparisons have been made to operating reactor data measured during startup tests and during normal power operation.

Validation of the cross section method is based on analysis of critical experiments, isotopic data, plant critical boron concentration data, and control rod worth measurement data such as that shown in Table 4.3-8.

Confirmatory critical experiments on burnable absorber rods are described in WCAP-7806 (Reference 42).

Group constants for tungsten GRCAs are generated using the PARAGON lattice code. Like PHOENIX-P, PARAGON is a two-dimensional, multigroup, transport-based lattice code capable of providing necessary data for PWR analysis and is approved by the NRC in Reference 69. WCAP-16943-P-A (Reference 70) contains a description of the nuclear methods for modeling tungsten and PARAGON benchmark results to Monte-Carlo simulations for assemblies containing tungsten GRCAs.

The PARAGON lattice code is also capable of generating all of the group constants generated by PHOENIX-P, and has been benchmarked and qualified to the same degree as PHOENIX-P. The NRC has approved the use of PARAGON as an alternative method for generating all macroscopic and microscopic group constants for uranium fueled cores (Reference 69). The primary difference between PARAGON and PHOENIX-P is that PARAGON uses Collision Probability theory with the interface current method to solve the integral transport equation. PARAGON also allows increased flexibility in modeling the exact assembly and pin cell geometry. The group constants generated by PARAGON are coupled to the spatial few-group code using the NEXUS nuclear data methodology (Reference 72).

4.3.3.3 Spatial Few-Group Diffusion Calculations

The 3D ANC code (see WCAP-10965-P-A, References 57 and 73) permits the introduction of advanced fuel designs with axial heterogeneities, such as axial blankets and part-length burnable absorbers, and allows such features to be modeled explicitly. The three dimensional nature of this code provides both radial and axial power distribution. For some applications, the updated version APOLLO (see WCAP-13524 Reference 60) of the PANDA code (see WCAP-7048-P-A Reference 35) may be used for axial calculations, and a two-dimensional collapse of 3D ANC that properly accounts for the three-dimensional features of the fuel may be used for X-Y calculations.

Spatial few group calculations are carried out to determine the critical boron concentrations and power distributions. The moderator coefficient is evaluated by varying the inlet temperature in the same kind of calculations as those used for power distribution and reactivity predictions.

Validation of the reactivity calculations is associated with validation of the group constants themselves, as discussed in subsection 4.3.3.2. Validation of the Doppler calculations is associated with the fuel temperature validation discussed in subsection 4.3.3.1. Validation of the moderator coefficient calculations is obtained by comparison with plant measurements at hot zero power conditions, similar to that shown in Table 4.3-9.

Axial calculations may be used in place of the full three-dimensional nodal model to determine differential control rod worth curves (reactivity versus rod insertion) and to demonstrate load follow capability. Group constants are obtained from the three-dimensional nodal model by flux-volume weighing on an axial slicewise basis. Radial bucklings are determined by varying parameters in the buckling model while forcing the one-dimensional model to reproduce the axial characteristics (axial offset, midplane power) of the three-dimensional model.

Validation of the spatial codes for calculating power distributions involves the use of in-core and ex-core detectors and is discussed in subsection 4.3.2.2.7.

As discussed in subsection 4.3.3.2, calculation-measurement comparisons have been made to operating reactor data measured during startup tests and during normal power operation. These comparisons include a variety of core geometries and fuel loading patterns, and incorporate a reasonable extreme range of fuel enrichment, burnable absorber loading, and cycle burnup. Qualification data identified in References 40, 69, and 72 indicate small mean and standard deviations relative to measurement which are equal to or less than those found in previous reviews of similar or parallel approved methodologies. For the reload designs the spatial codes described above, other NRC approved codes, or both are used.

4.3.4 Combined License Information

The Combined License information requested in this subsection has been completely addressed in APP-GW-GLR-059 (Reference 64), and the applicable changes have been incorporated into the DCD. No additional work is required by the Combined License applicant to address the Combined License information requested in this subsection.

The following words represent the original Combined License Information Item commitment, which has been addressed as discussed above:

> Combined License applicants referencing the AP1000 certified design will address changes to the reference design of the fuel, burnable absorber rods, rod cluster control assemblies, or initial core design from that presented in the DCD.

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