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Revision 1

Boiling Water Reactor Licensing Methodology Summary

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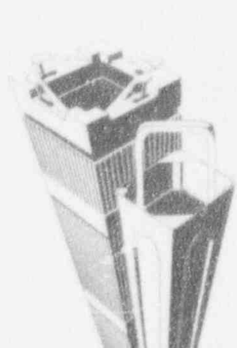

Siemens Power Corporation

Nuclear Division

Siemens Power Corporation - Nuclear Division

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Boiling Water Reactor
Licensing Methodology Summary

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Product Licensing

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1.0 INTRODUCTION

The introduction of nuclear fuel fabricated by Siemens Power Corporation - Nuclear Division (SPC) into the core of a boiling water reactor (BWR) requires assurance that the reactor will continue to meet accepted safety criteria during normal operation and accident conditions. Additionally, assurance must be provided that SPC-fabricated fuel is compatible with the existing fuel in the reactor core. In providing this assurance, SPC performs analyses for normal operations, anticipated operational occurrences, and postulated accidents which confirm or modify operating procedures, setpoints, and limits.

The results of theses analyses are provided to each customer in a report which follows a standard, NRC-approved, format.(43) The methodology used for these analyses is described in licensing topical reports issued by SPC and approved by the NRC. A complete bibliography of these reports is provided in Appendix A of this report. This report presents generic information covering the fuel design and analysis of BWRs for which SPC supplies reload fuel. This report also describes the application of the SPC BWR methodology for generic and plant specific analyses. In some cases, the design basis analyses for a plant, as documented in the FSAR, may differ from the approved SPC methodology. For these special cases, SPC will continue to support the original design basis as documented in the FSAR for that plant.

1.1 Fuel Assembly Design Objectives

SPC builds fuel assemblies to specific design criteria in order to assure that:

- The fuel assembly shall not fail as a result of normal operation and anticipated operational occurrences. The fuel assembly dimensions shall be designed to remain within operational tolerances and the functional capabilities of the fuel shall be established to either meet or exceed those assumed in the safety analysis.
- Fuel assembly damage shall never prevent control rod insertion when it is required.
- The number of fuel rod failures shall be conservatively estimated for postulated accidents.
- Fuel coolability shall always be maintained.

- The mechanical design of fuel assemblies shall be compatible with co-resident fuel and the reactor core internals.
- Fuel assemblies shall be designed to withstand the loads from in-plant handling and shipping.

The first four objectives are those cited in the Standard Review Plan.[1] The last two objectives assure structural integrity of the fuel and the compatibility of the fuel with existing reload fuel.

1.2 Reload Fuel Licensing Bases

The SPC design criteria are approved by the NRC[48] and are consistent with Chapter 4 of the Standard Review Plan. These criteria are chosen to provide assurance that all SPC BWR fuel designs will perform satisfactorily throughout their design lifetimes. Compliance with the design criteria is demonstrated by:

- Documenting the fuel system description and fuel assembly design drawings.
- Performing analyses with NRC-approved models and methods.
- Testing significant new design features with prototype testing and/or lead test assemblies prior to full reload implementation.
- Continued irradiation surveillance programs including post irradiation examinations to confirm fuel assembly performance.
- Using the NRC-approved QA procedures, QC inspection program, and design control requirements identified in the SPC Quality Assurance Manual.[2]

SPC BWR fuel designs which are demonstrated to meet the NRC-approved design criteria documented in Reference 48 do not need to be submitted to the NRC for explicit review and approval. Demonstration that a BWR fuel design meets the NRC-approved criteria is equivalent to formal NRC approval of the design.

As required for future designs, the design criteria presented in this report and Reference 48 will be evaluated and updated, as necessary. Any changes to the criteria will be submitted to the NRC for review and acceptance.

1.3 Fuel System Design Criteria

Chapter 4.2 of the Standard Review Plan contains criteria specified to provide assurance that the fuel system is not damaged as a result of normal operations or anticipated operational occurrences, that fuel system damage is never so severe as to prevent control rod insertion when it is required, that the number of fuel rod failures is not underestimated for postulated accidents, and that coolability is always maintained. These design criteria are necessary to meet the requirements of 10 CFR Part 50 (50.46; GDC 10, 27, and 35; Appendix K) and 10 CFR Part 100. The fuel system design criteria are summarized in Table 2.2 and discussed in further detail in Section 2.0 as well as in the referenced topical reports.

1.3.1 Fuel Rod

The detailed fuel rod design establishes such parameters as pellet diameter and density, cladding-pellet diametral gap, fission gas plenum size, and rod pre-pressurization level. The design also considers effects and physical properties of fuel rod components which vary with burnup. The integrity of the fuel rods is ensured by designing to prevent excessive fuel temperatures, excessive rod internal gas pressures, and excessive cladding stresses and strains. This end is achieved by designing the fuel rods to satisfy the design criteria during normal operation and anticipated operational occurrences over the fuel lifetime. For each design criterion, the performance of the most limiting fuel rod shall not exceed the specified limits.

1.3.2 Fuel System

Fuel system criteria are established to assure that fuel system dimensions remain within operational tolerances and that functional capabilities of the fuel assembly (system) are not reduced below those assumed in the safety analysis. The criteria apply for normal operation and for anticipated operational occurrences. The SPC criteria for the fuel system include those topics identified in the Standard Review Plan, as discussed in Section 2.0 and summarized in Table 2.2.

1.3.3 Fuel Coolability

To meet the requirements of General Design Criteria 27 and 35 as they relate to control rod insertability and core coolability for postulated accidents, fuel coolability criteria are established for all severe damage mechanisms. Coolability, or coolable geometry, has traditionally implied that the fuel assembly retains its rod-bundle geometry with adequate coolant channels to permit removal of residual heat. Reduction of coolability can result from cladding embrittlement, violent expulsion of fuel, generalized cladding melting, gross structural deformation, and extreme fuel rod ballooning.

1.4 Thermal and Hydraulic Design Criteria

Fuel designs are evaluated relative to the thermal and hydraulic design criteria to determine and provide thermal operating limits with acceptable margins of safety during normal reactor operation and anticipated operational occurrences. To the extent possible, these analyses are performed on a generic fuel design basis. Because of reactor and operating differences, however, many of the analyses supporting these thermal and hydraulic operating limits are performed on plant and cycle specific bases.

SPC uses NRC-approved methods and models[3,4,5] in the thermal and hydraulic design and analysis of new fuel designs and new fuel design features. In the event the proposed design features are determined to be outside the range of the methods and models, applicable documentation will be submitted to the NRC for review and approval.

1.5 Nuclear Design Analysis

The nuclear design analyses are subdivided into two parts: a nuclear fuel assembly design analysis and a core design analysis. The fuel bundle nuclear design analysis is assembly specific and changes only as features affecting the nuclear characteristics of the fuel change, i.e., rod enrichments, burnable absorber content, etc. The core nuclear design analysis is specific to the core configuration and changes on a cycle basis. Nuclear fuel and core analyses are performed using NRC-approved methodology[6] to assure that the new fuel assembly and/or design features meet the nuclear design criteria established for the fuel and core.

The fuel bundle nuclear design characteristics are considered for each SPC fuel bundle design added to the core. The key characteristics affecting the nuclear design analysis include the following items:

- Assembly average enrichment;
- Radial and axial enrichment distribution;
- Burnable absorber content and distribution; and
- Nature and location of non-fueled rods or water channels.

These key characteristics establish the fuel (local) and core power distributions and the kinetic parameters which are used in the thermal hydraulic, mechanical, and nuclear safety evaluations. The key neutronic design characteristics are selected such that fuel design limits are not exceeded during either normal operation or anticipated operational occurrences, and that the effects of postulated reactivity accidents will not cause significant damage to the reactor coolant pressure boundary or impair the capability to cool the core. These fuel assembly characteristics are evaluated on a reload cycle specific basis during the neutronic and thermal hydraulic safety analyses.

The core nuclear characteristics are evaluated during the core design analysis. These analyses include evaluation of the power distributions, kinetic parameters, control rod worths, etc. These characteristics are calculated for the reference core loading configuration for each operating cycle, as discussed further in Section 4.2.

1.6 Testing, Inspection, and Surveillance

The SPC testing and inspection requirements are essential elements in assuring conformance to the design criteria. The component parameters either directly demonstrate compliance with the design criteria or are input for the design calculations. Therefore, the components must be as specified.

The SPC Quality Control program provides assurance that the components satisfy the product specifications. The SPC Quality Assurance Manual[2] controls and maintains this program. The NRC has reviewed and accepted this manual as being in compliance with Appendix B of 10 CFR 50.[7]

The specific QC inspections performed by SPC include component parts, pellets, rods, and assemblies, as well as process control inspections. In addition, SPC reviews and overchecks inspections performed by vendors. These SPC and vendor inspections provide verification that the manufactured fuel is consistent with the fuel design.

New SPC fuel designs incorporate proven design features in combination with new design features which improve some fuel performance characteristics. SPC introduces new fuel designs through Lead Fuel Assembly programs which include surveillance of the in-reactor performance of the new design features.

The particulars of a Lead Fuel Assembly surveillance program depend on the specifics of the new fuel design features and are developed on a case specific basis in cooperation with the utility in whose reactor the Lead Fuel Assemblies are irradiated.

SPC can perform detailed visual inspection of fuel assemblies and make poolside measurements of the following parameters depending on the design changes being introduced:

- Rod Diameter Profilometry (Cladding Creepdown, Ovality, and Ridging)
- Oxide Thickness (Rods & Spacers)
- Cladding Wall Thinning
- Rod Length
- Fission Gas Release (Nondestructive and Destructive)
- Pellet Column Slump
- Pellet Column Gaps
- Axial Power Profile
- Isotope Redistribution
- Cladding Defects
- Rod Bow and Rod-to-Rod Spacing
- Channel Closure
- Crud Sampling (Chemical Composition)
- Spacer Spring Relaxation
- Assembly Length

Surveillance programs of the irradiated fuel provide confirmation of the design adequacy. SPC has performed extensive poolside examinations of irradiated fuel designs. These surveillance programs have confirmed the good performance of the SPC fuel. Post irradiation surveillance programs will continue to be an important part in assuring and confirming the adequacy of current and future SPC fuel designs.

1.7 Fuel Assembly Reconstitution

The application of SPC-approved design methodology to justify reinsertion into a reactor core irradiated fuel assemblies which have been reconstituted with replacement rods is described in Reference 49. Replacement rods can be fuel rods containing natural uranium pellets, water rods, and inert rods containing Zircaloy or stainless steel inserts.

Removal and replacement of suspected or known leaking fuel rods from irradiated assemblies has obvious advantages. Radiation levels in the plant are reduced with the removal of leaking fuel rods. Valuable technical information regarding fuel performance may be gathered through inspections. Where use of reconstituted fuel assemblies shows no significant impact on assembly and core performance, core loadings previously analyzed may be preserved and uranium that would otherwise be lost to energy production may be utilized.

A variety of replacement rod types may be used to reconstitute irradiated fuel assemblies depending on customer preference and application. In addition to fuel rods containing natural uranium, SPC uses either Zircaloy or stainless steel filled inert rods as well as water rods.

Replacement fuel rods containing natural uranium pellets have the advantage of more closely matching neutronic and mechanical characteristics of the remaining fuel rods in an irradiated fuel assembly. However, using fuel rods adds cost and complexity in manufacturing, transportation, and material accountability relative to using non-fuel replacement rods. In addition, a fuel replacement rod is not used in cases where there is a potential for failure of the replacement rod in subsequent operating cycles (e.g., spacer damage, etc.).

Zircaloy or stainless steel filled rods and water rods are used depending on customer preference. These types of replacement rods allow fast response to customer requirements since they can be easily manufactured and transported. Neutronic, thermal-hydraulic, and mechanical aspects of the reconstituted fuel assembly are reviewed for each replacement rod application. The review, performed with NRC-approved methodology, assures that design limits are applicable.

2.0 MECHANICAL DESIGN ANALYSES

2.1 Fuel System Damage

The design criteria for Fuel System Damage should not be exceeded during normal operations, including Anticipated Operational Occurrences (AOOs).

2.1.1 Stress

Design Criteria. The design criteria for evaluating the structural integrity of the fuel assemblies follow:

- Fuel assembly handling - The assembly must withstand dynamic axial loads approximately 2.5 times assembly weight.

- For all applied loads for normal operation and anticipated operational events - The fuel assembly component structural design criteria are established for the two primary material categories, austenitic stainless steels (tie plates) and Zircaloy (tie rods, grids, spacer capture rod tubes). The stress categories and strength theory for austenitic stainless steel presented in the ASME Boiler and Pressure Vessel Code, Section III, are used as a general guide.

  Steady state stress design limits are given in Table 2.1. Stress nomenclature is per the ASME Boiler and Pressure Vessel Code, Section III.

- Loads during postulated accidents - Deflection or failure of components shall not interfere with reactor shutdown or emergency cooling of the fuel rods.

Bases. In keeping with the GDC 10 SAFDLs, the fuel damage design criteria for cladding stress assure that fuel system dimensions remain within operational tolerances and that functional capabilities are not reduced below those assumed in the safety analysis. Conservative stress limits are derived from the ASME Boiler Code, Section III, Article III-2000;[8] and the specified 0.2% offset yield strength and ultimate strength for Zircaloy.

The structural integrity of the fuel assemblies is assured by setting design limits on stresses and deformations due to various handling, operational, and accident loads. These limits are applied to the design and evaluation of upper and lower tie plates, grid spacers, tie rods, spacer capture rod, water rods, water channels, fuel assembly cage, and springs where applicable. The allowable component stress limits are based on Appendix F of the ASME Boiler and Pressure Vessel Code, Section III, with some criteria derived from component tests. Cladding stress categories include the primary membrane and bending stresses, and the secondary stresses. The loadings considered are fluid pressure, internal gas pressure, thermal gradients, restrained mechanical bow, flow induced vibration, and spacer contact. Table 2.1 gives the ASME stress level criteria.

The stress calculations use conventional, elasticity theory equations. A general purpose, finite element stress analysis code such as ANSYS(9) is used to calculate the spacer spring contact stresses. The fuel assembly structural component stresses under faulted conditions are evaluated using primarily the criteria outlined in Appendix F of the ASME Boiler and Pressure Vessel Code, Section III.

The SPC analysis methods for calculating fuel rod cladding and assembly steady-state stresses are discussed and approved in References 10 and 11.

2.1.2 Strain

Design Criteria. The SPC design criteria for fuel rod cladding strain is that the transient-induced deformations must be less than 1% uniform cladding strain.

Bases. The design criteria for cladding strain is intended to preclude excessive cladding deformation and failure from normal operations and AOOs. SPC uses the NRC-approved RODEX2A code(12) to calculate steady-state cladding strain during normal operation. The RAMPEX(13) code is used by SPC to calculate cladding strain during transient operation.

2.1.3 Strain Fatigue

Design Criteria. The SPC design criteria for strain fatigue limits the cumulative fatigue usage factor to less than

Bases. Cycle loading associated with relatively large changes in power can cause cumulative damage which may eventually tend to fatigue failure. Therefore, SPC requires that the cladding not exceed a cumulative fatigue usage factor The fatigue usage factor is the number of expected cycles divided by the number of allowed cycles. The total cladding usage factor is the sum of the individual usage factors for each duty cycle.

The SPC methodology for determining strain fatigue is based on the use of the RAMPEX[13] code and the O'Donnell and Langer fatigue design curves.[14] The fatigue curves have been adjusted to incorporate the recommended safety factor of two on stress amplitude or 20 on number of cycles, whichever is more conservative. The RODEX2 code is used to provide initial steady-state conditions for SPC transient and accident analysis. Consequently, the RODEX2 code provides input to the RAMPEX code for each power change, and RAMPEX provides stress amplitudes for the various power cycles.

2.1.4 Fretting Wear

Design Criteria. The SPC design criteria for fretting wear requires that fuel rod failure due to fretting shall not occur.

Bases. SPC controls fretting wear by use of design features, such as a spacer spring dimple system, which assure that fuel rods are positively supported by the grid spacers throughout the expected irradiation period. Spacer grid spring systems are designed such that the

SPC performs fretting tests to verify consistent fretting performance for new spacer designs. Examination of a large number of irradiated BWR rods has substantiated the absence of fretting in SPC designs.

2.1.5 Oxidation and Crud Buildup

Design Criteria. There is no specific limit for oxide thickness or crud buildup. The effects of oxidation and crud buildup are included in the thermal and rod internal gas pressure analysis.

Bases. The SPC fuel design basis for cladding corrosion and crud buildup is to prevent 1) significant degradation of the cladding strength, and 2) unacceptable temperature increases. Cladding corrosion reduces cladding wall thickness and results in less cladding load carrying capacity. At normal light water reactor operating conditions, this mechanism is not limiting except under unusual conditions where high cladding temperatures greatly accelerate the corrosion rate. Because of the thermal resistance of corrosion and crud layers, formation of these products on the cladding results in an elevation of temperature within the fuel as well as the cladding.

There is no specific limit for crud buildup. However, SPC fuel performance codes[12,25] reviewed and approved by the NRC include the crud buildup in the fuel performance predictions. That is, the crud and oxidation models are a part of the approved models and therefore impact the temperature calculation.

SPC data show that even at higher exposures and residence times, cladding oxide thickness is relatively low. Mechanical properties of the cladding are not significantly affected by thin oxide or crud layers. For the thermal and rod internal gas pressure analyses, the effect of oxidation is included. For steady-state strain, transient strain and cyclic stress, the effect of wall thinning is insignificant since cladding deformation is strain dependent. That is, the change in cladding diameter during a power change is primarily determined by the change in the pellet diameter since pellet-cladding contact occurs at higher exposures. For the cladding end-of-life stress analysis, the wall thickness is reduced consistent with the peak oxide thickness.

2.1.6 Rod Bowing

Design Criteria. The SPC design criteria for rod bowing is that

Bases. Differential expansion between the fuel rods, and lateral thermal and flux gradients can lead to lateral creep bow of the rods in the spans between spacer grids. This lateral creep bow alters the pitch between rods and may affect the peaking and local heat transfer.

SPC has established a rod-to-rod clearance closure limit[11] below which a penalty is addressed on the minimum critical power ratio (MCPR) and above which no reduction in MCPR is necessary.

SPC uses NRC-approved models[11] which are based on empirical data to calculate minimum EOL rod to rod spacing. The potential effect of this rod bow on thermal margin is negligible. Rod bow at extended burnup does not affect thermal margins due to the lower powers achieved at high exposure.

2.1.7 Axial Growth

Design Criteria. SPC requires that the fuel assembly be compatible with the channel throughout the fuel assembly lifetime. In addition, SPC requires the fuel rods and other assembly components to maintain clearances and engagements in the fuel assembly structure throughout the lifetime of the fuel.

Bases. SPC evaluates fuel channel-fuel assembly differential growth to assure that the fuel channel to lower tie plate engagement is maintained to the design burnup. Another concern for BWR fuel assemblies is to maintain engagement between the fuel rod end cap shank and the assembly tie plates, i.e., to prevent fuel rod disengagement from the tie plates. The change in BWR fuel rod-to-tie plate engagement (and possible disengagement) is due to the growth rate of the tie rods that connect the bottom and top tie plates being greater than the growth rate of the fuel rods.

The SPC analysis method for evaluating rod-to-tie plate engagement is based on a statistical upper bound of measured differential rod-to-tie plate growth from both 8x8 and 9x9 designs.[11,15,25]

2.1.8 Rod Internal Pressure

Design Criteria. SPC limits maximum fuel rod internal pressure relative to system pressure. In addition, SPC requires that when fuel rod pressure exceeds system pressure, the pellet-clad gap has to remain closed if it is already closed or that it should not tend to open for steady state or increasing power operations.

Bases. Rod internal pressure is limited to prevent unstable thermal behavior and to maintain the integrity of the cladding. Outward circumferential creep which may cause an increase in pellet-to-cladding gap must be prevented since it would lead to higher fuel temperature and higher fission gas release. The maximum internal pressure is also limited to protect against embrittlement of the cladding caused by hydride reorientation during cooldown and depressurization conditions. A proprietary limit above system pressure has been justified by SPC in Reference 15.

2.1.9 Assembly Liftoff

Design Criteria. SPC requires that the assembly not levitate from hydraulic or accident loads.

Bases. Levitation of a fuel assembly could result in the assembly becoming disengaged from the fuel support and interfering with control rod movement. For normal operation, including AOOs, the submerged fuel assembly weight, including the channel, must be greater than the hydraulic loads. The criteria is applicable to both cold and hot conditions and uses the Technical Specification limits on total core flow. For accident conditions, the normal hydraulic loads plus additional accident loads shall not cause the assembly to become disengaged from the fuel support. This assures that control blade insertion is not impaired.

2.1.10 Fuel Assembly Handling

Design Criteria. The assembly design must withstand all normal axial loads from shipping and fuel handling operations without permanent deformation.

Bases. SPC uses either a stress analysis or testing to demonstrate compliance. The analysis or test uses an axial load of 2.5 times the static fuel assembly weight. At this load, the fuel assembly structural components must not show any yielding.

The rod plenum spring also has design criteria associated with handling requirements.

The component drawing specifies the fabricated cold spring force.

2.1.11 Miscellaneous Component Criteria

2.1.11.1 Compression Spring Forces

Design Criteria.

Bases. The compression springs must support the weight of the upper tie plate and channel throughout the design life of the fuel. Therefore, SPC has a requirement on the minimum compression spring force.

2.1.11.2 Lower Tie Plate Seal Spring

Design Criteria. The seal accommodates the channel deformation to limit the leak rate of coolant between the lower tie plate and channel wall.

Bases. The lower tie plate seal spring limits the leak rate of coolant between the lower tie plate and the channel wall. The seal shall have adequate corrosion resistance and be able to withstand the operating stresses without yielding. The design also considers the differential axial growth between the channel and the bundle. Flow testing of prototypic components verifies the leakage rate and fretting resistance. A stress analysis provides the seal stresses.

2.2 Fuel Rod Failure

The fuel rod failure criteria and bases cover normal operation conditions, including AOOs, and postulated accidents. When the fuel rod failure criteria are applied in normal operation, including AOOs, they are used as limits (Specified Acceptable Fuel Design Limits) since fuel failure under those conditions should not occur according to GDC 10.[16] When the criteria are used for postulated accidents, fuel failures are permitted, but they must be accounted for in the dose calculations required by 10 CFR 100.[17]

2.2.1 Internal Hydriding

Design Criteria. SPC limits internal hydriding by imposing a fabrication limit for total hydrogen in the fuel pellets of less than 2.0 ppm.

Bases. The absorption of hydrogen by the cladding can result in cladding failure due to reduced ductility and formation of hydride platelets. Hydriding, as a cladding failure mechanism, is precluded by controlling the level of moisture and other hydrogenous impurities during fuel pellet fabrication. SPC assures that the hydrogen concentration criteria is met through careful moisture control during fuel fabrication.[15]

2.2.2 Cladding Collapse

Design Criteria. Creep collapse of the cladding is avoided in the SPC fuel system design by

Bases. If axial gaps in the fuel pellet column were to occur due to handling, shipping, or fuel densification, the cladding would have the potential of collapsing into the gap. Because of the large local strains that would result from the collapse, the cladding is assumed to fail. Creep collapse of the cladding and the subsequent potential for fuel failure is avoided in the SPC fuel system design by

SPC uses the NRC-approved RODEX2A[12] and COLAPX[18] codes to predict creep collapse. The RODEX2A code is used to provide initial in-reactor fuel rod conditions to COLAPX, e.g., radial fuel-cladding gap size, fill gas pressure, and cladding temperatures. The COLAPX code calculates ovality changes and creep deformation of the cladding as a function of time.

2.2.3 Overheating of Cladding

Design Criteria. The SPC design basis to preclude fuel rod cladding overheating is that 99.9% of the fuel rods shall not experience transition boiling.

Bases. It has been traditional practice to assume that fuel failures will occur if the thermal margin criteria is violated. Thermal margin is stated in terms of the minimum value of the critical power ratio (CPR) for the most limiting fuel assembly in the core. Prevention of potential fuel failure from overheating of the cladding is accomplished by minimizing the probability of exceeding thermal margin limits on limiting fuel rods during normal operation and anticipated operational occurrences. SPC confirms compliance with this criteria as part of the reload thermal hydraulics analysis, as discussed in Section 3.2 of this report. Experimentally based critical heat flux correlations which have been accepted by the NRC are used (see Section 3.2).

2.2.4 Overheating of Fuel Pellets

Design Criteria. Fuel failure from overheating of the fuel pellets is not allowed. The centerline temperature of the fuel pellets must remain below melting during normal operations and AOOs.

Bases. SPC establishes steady state and transient design LHGR limits for each fuel system which protect against centerline melting. Operation within these LHGR limits prevents centerline melting during normal operation and anticipated operational occurrences throughout the design lifetime of the fuel.

SPC utilizes a correlation for the fuel melting point that accounts for the effect of burnup and gadolinia content. This fuel melting limit has been reviewed and approved by the NRC[15] with respect to application to fuel and gadolinia bearing fuel at extended burnup levels because the limit conservatively accounts for the decrease in fuel melting point with increasing burnup.

SPC uses the RODEX2A computer code[12] to calculate maximum possible fuel centerline temperature for normal operations. Conservative LHGR power histories are used to perform the centerline temperature calculations. For AOOs, SPC uses the RODEX2A code to calculate maximum possible fuel centerline temperatures with an LHGR history which is higher than the steady-state LHGR history used for normal operations by a conservative factor.

2.2.5 Pellet/Cladding Interaction

Design Criteria. The Standard Review Plan[1] does not contain an explicit criteria for pellet/cladding interaction. However, it does present two related criteria. The first one is that transient-induced deformations must be less than 1% uniform cladding strain. And, the second one is that fuel melting cannot occur. SPC requires that: 1) transient-induced deformations must be less than 1% uniform cladding strain, and 2) fuel centerline melting cannot occur.

Bases. The cladding strain requirement is addressed in Section 2.1.2 of this document. The centerline temperature requirement is addressed in Section 2.2.4 of this document.

2.2.6 Cladding Rupture

Design Criteria. 10 CFR 50 Appendix K[19] requires that cladding rupture must not be underestimated when analyzing a loss of coolant accident.

Bases. Zircaloy cladding will burst (rupture) under certain combinations of temperature, heating rate, and differential pressure conditions during a LOCA. While there are no specific design criteria in the Standard Review Plan[1] associated with cladding rupture, the requirements of Appendix K to 10 CFR 50 must be met as those requirements relate to the incidence of rupture during a LOCA; therefore, a rupture temperature correlation must be used during the LOCA emergency core cooling system (ECCS) analysis.

SPC includes the effects of cladding rupture as an integral part of the ECCS evaluation model. SPC uses the cladding ballooning and rupture models presented in NUREG-0630[20] for cladding rupture evaluation. These models are described in XN-NF-82-07(P)(A), Revision 1,[21] which has been reviewed by the NRC and found acceptable for use in LOCA analyses. The link between the cladding deformation and rupture models, and the LOCA ECCS analysis is described in Reference 22.

2.2.7 Fuel Rod Mechanical Fracturing

Design Criteria. SPC limits the combined stresses from postulated accidents to the stresses given in the ASME Code, Section III, Appendix F[8] for faulted conditions.

Bases. A mechanical fracture refers to a defect in a fuel rod caused by an externally applied force, such as a hydraulic load or a load derived from core plate motion induced by seismic/LOCA events. The design bases and criteria for mechanical fracturing of SPC BWR reload fuel are presented in Reference 23, which describes SPC's LOCA-seismic structural response analysis. LOCA-seismic structural response analysis covering SPC's 9x9 fuel designs is presented in Reference 24. The design basis is that the channeled fuel assemblies must withstand the external loads due to earthquake and postulated pipe breaks without fracturing the fuel rod cladding. The stresses, due to postulated accidents in combination with normal steady-state fuel rod stresses, should not exceed the stress limits given in References 23 and 24. The allowable stresses are derived from the ASME Boiler and Pressure Vessel Code, Section III, Appendix F, for faulted conditions. The design limits have been reviewed and approved by the NRC.

2.2.8 Fuel Densification and Swelling

Design Criteria. Fuel densification and swelling are limited by the design criteria specified for fuel temperature, cladding strain, cladding collapse, and internal pressure criteria.

Bases. SPC uses the NRC reviewed and accepted densification and swelling models in the fuel performance codes.[12,25]

2.3 BWR Fuel Coolability

For accidents in which severe fuel damage might occur, core coolability and the capability to insert control blades are essential. Normal operation or anticipated operational occurrences must remain within the thermal margin criteria. Chapter 4.2 of the Standard Review Plan[1] provides several specific areas important to the coolability and the capability of control blade insertion. The sections below discuss these areas.

2.3.1 Fragmentation of Embrittled Cladding

Design Criteria. SPC ECCS evaluations meet the 10 CFR 50.46[26] limits of 2200 °F peak cladding temperature, local and core-wide oxidation, and long term coolability.

Bases. The requirements on cladding embrittlement relate to the LOCA requirements of 10 CFR 50.46. The principal cause of cladding embrittlement during severe accidents such as LOCA is the high cladding temperatures that result in severe cladding oxidation.

The models to compute the temperatures and oxidation are those prescribed by Appendix K of 10 CFR 50.[19] The SPC methodology for evaluating cladding oxidation and embrittlement during a LOCA is included in the approved topical reports for LOCA-ECCS analysis.[22,27] The LOCA analysis is performed on a plant specific basis.

2.3.2 Violent Expulsion of Fuel

Design Criteria. SPC limits the radially-averaged enthalpy deposition at the hottest axial location to 280 cal/gm for severe reactivity initiated accidents.

Bases. In a severe reactivity initiated accident (RIA), large and rapid deposition of energy in the fuel could result in melting, fragmentation, and dispersal of the fuel. SPC has adopted the guideline in Regulatory Guide 1.77[28] and the Standard Review Plan[1] that restricts the radially-averaged energy deposition. SPC uses the 280 cal/gm radially-averaged enthalpy deposition at the hottest axial location as the design criteria.

The limiting RIA for SPC fuel in a boiling water reactor is the control rod drop accident. SPC calculates the maximum radially averaged enthalpy for the CRDA for each reload core in order to assure that the maximum calculated enthalpy is below the 280 cal/gm limit. The SPC control rod drop calculation methodology approved by the NRC is described in Reference 29. The parameterized SPC control rod drop methodology determines maximum deposited enthalpy as a function of dropped rod worth, effective delayed neutron fraction, Doppler coefficient, and four-bundle local peaking factor.

The CRDA analysis is not part of the normal fuel assembly mechanical analysis but is part of the cycle specific safety analysis performed for each BWR.

2.3.3 Cladding Ballooning

Design Criteria. There are no specific design limits associated with cladding ballooning, other than a 10 CFR 50 Appendix K[19] requirement that the degree of swelling not be underestimated.

Bases. Zircaloy cladding will balloon (swell) under certain combinations of temperature, heat rate, and stress during a LOCA. Cladding ballooning can result in flow blockage; therefore, the LOCA analysis must consider the cladding ballooning impacts on the flow.

The SPC cladding ballooning model is an integral part of the cladding rupture temperature model for the LOCA ECCS analysis. The cladding rupture temperature model is addressed in Reference 21. SPC, in Reference 21, has adopted the NUREG-0630[20] data base and modeling, which specifies a method acceptable to the NRC for treating cladding swelling and rupture during a LOCA. These models have been approved by the NRC for extended burnup levels.[15]

The RODEX2 fuel performance code[25] is used to provide burnup dependent input to the LOCA analysis, e.g., stored energy and rod pressures, that are a function of the initial steady-state operation of the fuel. This initial steady-state fuel condition is also important to cladding ballooning.

2.3.4 Fuel Assembly Structural Damage from External Forces

Design Criteria. The SPC design criteria for fuel assembly structural damage from external forces are covered in Sections 2.1.1, Stress, 2.1.9, Assembly Liftoff, and 2.2.7, Fuel Rod Mechanical Fracture.

Bases. Earthquakes and postulated pipe breaks in the reactor coolant system would result in external forces on the fuel assembly. The Standard Review Plan[1] states that fuel system coolability should be maintained and that damage should not be so severe as to prevent control blade insertion when required during these accidents. The SPC design basis is that the fuel assembly will maintain a geometry that is capable of being cooled under the worst case accident Condition IV event and that system damage is never so severe as to prevent control blade insertions. SPC assures these design bases are met by placing ASME design limits on the stresses that critical fuel assembly components can experience. These limits have been approved for SPC 8x8 and 9x9 fuel assemblies in References 23 and 24, respectively.

Table 2.1

Steady State Stress Design Limits*

	Stress Intensity Limits**	
	Yield Strength (σ_y)	Ultimate Tensile Strength (σ_u)
General Primary Membrane Stress	2/3 σ_y	1/3 σ_u
Primary Membrane Plus Primary Bending Stress	1.0 σ_y	1/2 σ_u
Primary Plus Secondary Stress	2.0 σ_y	1.0 σ_u

* Characteristics of the stress categories are defined as follows:

a) Primary stress is a stress developed by the imposed loading which is necessary to satisfy the laws of equilibrium between external and internal forces and moments. The basic characteristic of a primary stress is that it is not self-limiting. If a primary stress exceeds the yield strength of the material through the entire thickness, the prevention of failure is entirely dependent on the strain-hardening properties of the material.

b) Secondary stress is a stress developed by the self-constraint of a structure. It must satisfy an imposed strain pattern rather than being in equilibrium with an external load. The basic characteristic of a secondary stress is that it is self-limiting. Local yielding and minor distortions can satisfy the discontinuity conditions due to thermal expansions which cause the stress to occur.

** The stress intensity is defined as twice the maximum shear stress and is equal to the largest algebraic difference between any two of the three principal stresses.

Table 2.2

Summary Description of Fuel System Design Criteria

Section	Description	Generic Design Criteria
2.1	Fuel System Criteria	
2.1.1	Stress, Strain, or Loading Limits on Assembly Components	Table 2.1 Steady State (See 2.3.4 below for accident)
2.1.2	Cladding Strain	1% strain
2.1.3	Fatigue	Cumulative usage factor
2.1.4	Fretting Wear	No significant fretting wear
2.1.5	Oxidation, Hydriding and Crud Buildup	Acceptable metal loss due to oxidation
2.1.6	Rod Bow	Protect Thermal Limits
2.1.7	Axial Irradiation Growth	Seal Spring compatible with channel Maintain end cap engagements in UTP
2.1.8	Rod Internal Pressure	Radial gap does not open, no hydride reorientation
2.1.9	Assembly Liftoff	Maintain assembly engagement in core support piece and maintain positive holddown force
2.1.10	Fuel Assembly Handling	Assembly withstand 2.5 times static weight as axial load
2.1.11	Miscellaneous Components	
2.1.11.1	Compression Spring Force	
2.1.11.2	Lower Tie Plate Seal Spring	Accommodate channel deformation adequate corrosion resistance withstand operating stresses

Table 2.2

Summary Description of Fuel System Design Criteria (Continued)

Section	Description	Generic Design Criteria
2.2	Fuel Rod Criteria	
2.2.1	Internal Hydriding	<2 ppm H_2
2.2.2	Cladding Collapse	
2.2.3	Overheating of Cladding	99.9% of rods not to exceed CHF
2.2.4	Overheating of Fuel Pellets	No centerline melting
2.2.5	Pellet/Cladding Interaction	1% strain No fuel melting
2.2.6	Cladding Rupture	Not underestimated during LOCA and used in determination of 10 CFR 50.46 criteria(26)
2.2.7	Mechanical Fracturing Limits	ASME Section III, Appendix F(8)
2.2.8	Densification and Swelling	Included in 2.1.8, 2.2.4, and 2.2.5
2.3	Fuel Coolability	
2.3.1	Cladding Embrittlement	Included in LOCA Analysis
2.3.2	Violent Expulsion of Fuel	<280 cal/gram
2.3.3	Fuel Ballooning	Consider impact on flow blockage in LOCA analysis
2.3.4	Structural Deformations	Coolable geometry, control rod insertability

3.0 THERMAL AND HYDRAULIC DESIGN ANALYSES

Thermal-hydraulic analyses of the fuel and core are performed to verify that design criteria are satisfied and to establish an appropriate value for the MCPR fuel cladding integrity safety limit.

3.1 Hydraulic Compatibility

Design Criteria. The hydraulic flow resistance of the reload fuel assemblies shall be sufficiently similar to existing fuel in the reactor that there is no significant unplanned impact on total core flow or the flow distribution among assemblies in the core.

Bases. The Standard Review Plan[1] does not contain an explicit criterion for fuel assembly hydraulic compatibility. However, flow differences between assembly types in a mixed core need to be accounted for in assuring that all design criteria are satisfied.

The component hydraulic resistances in the reactor core are determined by a combination of both analytical techniques and experimental data.

The SPC thermal-hydraulic methodology implicitly includes the impact of assembly differences on the individual assembly flow. The overall criterion for acceptability is that individual fuel types must be in compliance with the thermal hydraulic limits.

The purpose of these evaluations is to better define the core stability behavior with this mismatch in flow. The MCPR performance remains protected by the compliance with the safety and operating limits.

3.2 Thermal Margin Performance

Design Criteria. The fuel design shall fall within the limits of applicability of the approved critical heat flux (CHF) correlation. New fuel assembly designs and/or changes in existing assembly designs shall minimize the likelihood of boiling transition during normal reactor operation and AOOs. The applicable critical power correlation will be used to determine the operating limits and for consistency be used to monitor the core.

Bases. SPC fuel and reload cores are designed such that operation consistent with Technical Specification limits will result in <0.1% of the rods experiencing boiling transition during AOOs. An NRC-approved CHF correlation is used by SPC to determine operating and safety limits during the design of a reload core, and, for consistency, the same CHF correlation is used to monitor the core during operation.

Operation of a BWR requires protection against fuel damage during normal reactor operation and AOOs. A rapid decrease in heat removal capacity associated with boiling transition can potentially result in high transient temperatures in the cladding, which may cause cladding degradation and a loss of the fuel rod integrity. Protection of the fuel against boiling transition assures that such degradation is avoided. This protection is accomplished by determining the operating limit minimum critical power ratio (OLMCPR) for each fuel bundle in the reactor core for each cycle.

The SPC approach to thermal limits analysis, THERMEX, is described in Reference 3. This THERMEX thermal limits methodology consists of a series of related analyses which establish an operating limit minimum critical power ratio (OLMCPR). The OLMCPR is determined from two calculated values, the safety limit MCPR (SLMCPR) and the limiting transient ΔCPR. The overall methodology is comprised of four major segments, which are 1) reactor core hydraulic methodology, 2) an NRC-approved critical power correlation, 3) plant transient simulation methodology, and 4) critical power methodology.

SPC fuel assembly pressure drop methodology is presented in Reference 31. The methodology addresses part of the calculational method used by SPC to determine assembly pressure drop that is used to calculate assembly flows for a BWR core. The pressure drop methodology report identifies the constitutive relationships to determine void fraction and two-phase pressure losses which are in turn used as input to the calculation of the assembly pressure drop by the XCOBRA computer code.[3]

The calculation of the fuel assembly critical power performance is established by means of an empirical correlation based on results of boiling transition test programs. SPC currently employs two NRC-approved critical power correlations. The ANFB Critical Power Correlation is described in Reference 32. The older XN-3 Critical Power Correlation is described in References 33 and 34.

The methodology and computer codes for SPC BWR plant transient analyses are described in detail in References 4 and 5. This plant transient methodology is supplemented by the NRC-approved XCOBRA-T code[37,38] and the COTRANSA2 code.[39] The COTRANSA2 code is used to calculated BWR system behavior for steady-state and transient conditions. This behavior is then used to provide input to the XCOBRA-T and XCOBRA codes, from which critical power ratios are determined for limiting transients.

In the generation of the limiting transient ΔCPR in the THERMEX methodology, consideration may be given to the statistical convolution of uncertainties associated with the calculation of the thermal margin. The statistical uncertainty analysis methodology, GSUAM, is described on a generic basis in Reference 40. Each plant specific GSUAM application must contain the variances and distributions of the predictor variables used in the response surface fitting and sufficient data must be presented to identify the mean and statistical variation of each predictor variable. All plant parameters not treated statistically and any predictor variable which is eliminated from the response surface fitting must be set at their limiting value.

The critical power ratio methodology is the approach used by SPC to determine the margin-to-thermal limits for boiling water reactors. The SPC critical power methodology for boiling water reactors is presented in References 41 and 42. The Reference 41 topical report addresses the critical power methodology associated with the XN-3 CPR correlation while the Reference 42 topical addresses the more recent ANFB correlation.

The ANFB-based topical report[42] provides the basis for the SPC methodology for determining the operating safety limit for minimum critical power (SLMCPR) which ensures that 99.9% of the fuel rods are protected from boiling transition. This determination is carried out by a series of Monte Carlo calculations in which the variables affecting the probability of boiling transition are varied randomly and the total number of rods experiencing boiling transition is determined for each Monte Carlo trial.

The methodology and component uncertainties needed to calculate power distribution uncertainty are given in Reference 6 and the assembly flow calculational uncertainty is given in the Reference 42 topical report. The SPC methodology for analysis of assembly channel bow effects is also given in Reference 42. Figure 3.1 shows the relationship between the SPC transient and critical power methodologies.

The THERMEX analytical process requires the input of certain external data, a portion of which may be plant-dependent. The specific input sources are described Reference 3.

3.3 Fuel Centerline Temperature

Design Criteria. Fuel design and operation shall be such that fuel centerline melting is not projected for normal operation and AOOs.

Bases. This design criteria is addressed during the fuel type specific mechanical design analysis. The bases is discussed in Section 2.2.4 of this document.

3.4 Rod Bow

Design Criteria. The anticipated magnitude of fuel rod bowing under irradiation shall be accounted for in establishing thermal margins requirements.

Bases. The bases for rod bow are discussed in Section 2.1.6 of this document. Rod bow magnitude is determined during the fuel type specific mechanical design analyses. The need for a thermal margin rod bow penalty is evaluated on a plant and cycle specific basis. Post-irradiation examinations of BWR fuel fabricated by SPC show that the magnitude of fuel rod bowing is small and the potential effect of this bow on thermal margins is negligible. Rod bow at extended burnups does not affect thermal margins due to the lower powers achieved by high exposure assemblies.

3.5 Bypass Flow

Design Criteria.

Bases. The Standard Review Plan[1] does not contain an explicit criterion for fuel assembly bypass flow characteristics. However, significant changes in bypass region flow may alter the response characteristics of the incore neutron detectors. In order to avoid altering the incore neutron detector response characteristics, SPC evaluates bypass flow fraction on a plant and cycle specific basis

Table 3.1

Summary Description of Thermal and Hydraulic Design Criteria

Section	Description	Generic Design Criteria
3.1	Hydraulic Compatibility	Hydraulic flow resistance similar to resident fuel assemblies
3.2	Thermal Margin Performance	<0.1% for rods in boiling transition
3.3	Fuel Centerline Temperature	No centerline melting
3.4	Rod Bow	Protect thermal limits
3.5	Bypass Flow	

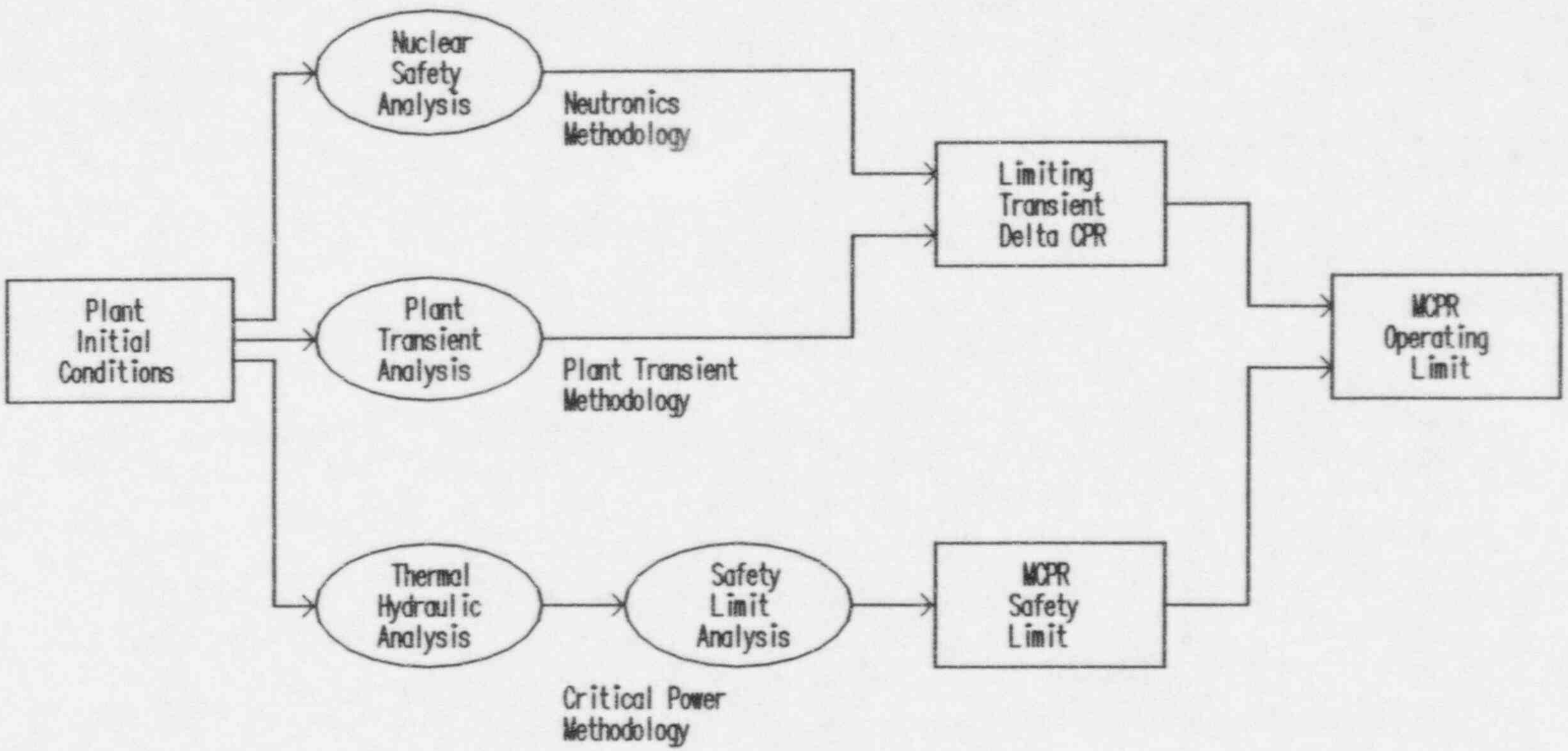

Figure 3.1

Calculational Flow Used by the THERMEX Thermal Limits Methodology

4.0 NUCLEAR DESIGN AND ANALYSES

4.1 Nuclear Design and Analysis Methodology

SPC's NRC-approved nuclear design and analysis methodology is centered around five computer codes:[29] 1) XFYRE - fuel assembly depletion model, 2) XTGBWR - core simulator model, 3) COTRAN - reactor kinetics model, 4) XDT - multi-group diffusion theory, and 5) XMC - Monte Carlo neutron diffusion. In Reference 6, SPC provided a basis for, and the NRC approved, the direct replacement of the fuel assembly depletion model and core simulator model with the CASMO-3G and MICROBURN-B codes, respectively. The SPC BWR neutronic methods for design and analysis[29] provide NRC-approved methodology for calculation of steady-state core parameters, evaluation of control rod drop accidents, analysis of fuel assembly mislocation, evaluation of control rod withdrawal errors and determination of neutronic reactivity parameters.

4.2 Nuclear Design Analysis

The nuclear design analyses are divided into two parts: a nuclear fuel assembly design analysis and a core design analysis. The design analysis should demonstrate operating margin to design limits, including MCPR, maximum average planar linear heat generation rate (MAPLHGR), and LHGR.

4.2.1 Fuel Rod Power History

Design Criteria. The nuclear design analysis must be consistent with the exposure dependent LHGR limit established during the mechanical design analysis for each fuel assembly design.

Two LHGR limits are established for each fuel design. One is a steady state limit, the other a transient limit. Both limits are a function of fuel planar burnup. The transient LHGR design limit protects strain and fuel overheating design criteria discussed in Sections 2.1.2 and 2.2.4.

The design margin between the steady state and transient LHGR limits is sufficient to account for increases in the LHGR during transients.

Bases. An exposure dependent LHGR limit is established for each fuel assembly design as part of the mechanical design analysis.[10,15] The LHGR limit bounds the power history established to perform the mechanical analyses. Therefore, operation of the fuel assembly within the LHGR limit is necessary to ensure that the power history assumption used in the mechanical design analyses remains valid. The specific mechanical design criteria are discussed in Section 2.0 of this document. During the nuclear design process, and in the reactor, the fuel is required to operate within this power history used to perform the mechanical analyses.

Exposure dependent Linear Heat Generation Rate (LHGR) limits are established for each fuel design. These limits are established based on nuclear design analyses and the fuel design criteria established in Section 2.0. Specifically, conservative power histories are generated based on proposed LHGR limits.

This assumption leads to conservative estimates of fission gas release.

4.2.2 Kinetics Parameters

Design Criteria. The design criteria for the reactivity coefficients are as follows:

- Void Reactivity Coefficient due to boiling in the active channel shall be negative;
- Doppler Coefficient shall be negative at all operating conditions;
- Power Coefficient shall be negative at all operating conditions.

Bases. Design of fuel assemblies such that the less moderation and/or higher temperatures reduce the reactivity of the core results in an automatic shutdown mechanism. Thus, prompt critical reactivity insertion events such as the control rod drop accident have an inherent shutdown mechanism. SPC calculates the reactivity coefficients on a plant and cycle specific basis through application of the standard neutronics design and analysis methodology.[6,29]

4.2.3 Stability

Design Criteria. New fuel designs and new fuel design features must be stable (core decay ratio <1.0) and exhibit channel decay ratio characteristics equivalent to existing SPC fuel designs.

Bases. Determination of the effect of all fuel designs and design features on core stability is made on a cycle specific basis. Associated with these licensing calculations is confirmation of existing exclusion regions or redefinition of the regions, as necessary.

SPC uses the NRC-approved STAIF code[30] for stability evaluations. STAIF is a frequency domain code that simulates the dynamics of a boiling water reactor. STAIF's main output is a series of transfer functions that define the linear dynamic behavior of: 1) the channel thermal-hydraulics, 2) the fundamental-mode coupled neutronics and thermal-hydraulics, and 3) the subcritical-modes coupled neutronics and thermal-hydraulics. STAIF estimates the decay ratios for the above three modes of oscillation using a mathematical procedure that applies to the computed transfer functions.

SPC will confirm that the stability performance of a new BWR fuel design is equivalent to or better than that of an approved SPC fuel design. The stability performance will be demonstrated by comparative analytic stability calculations for both the new fuel design and the existing approved SPC designs using the approved stability analysis code.

Because the stability performance of fuel is dominated by the power distribution, and to a lesser extent by fuel design variations, it is necessary to establish consistent boundary conditions when comparing for fuel design variations. Therefore, SPC will use the same power distribution and full core equilibrium cycle loading pattern for the fuel design comparison. The neutronic characteristics will remain unchanged with the exception of the void coefficient, which is directly dependent on the fuel design. Fuel mechanical design parameters, e.g., component dimensions, spacer loss coefficients, etc. will be input for each fuel design in the comparative analysis.

The BWR Owners Group is actively developing a generic resolution of the stability concerns. As the BWROG results become available and accepted by the NRC, and as they are implemented by the utilities on an individual basis, the SPC cycle specific stability determinations will likely be modified. These modifications will be included in the reactor cycle specific evaluations and are not expected to affect the comparative fuel design related stability performance.

4.2.4 Core Reactivity Control

Design Criteria. The design of the assembly shall be such that the Technical Specification shutdown margin will be maintained. Specifically, the assemblies and the core must be designed to remain subcritical by the Technical Specification margin with the highest reactivity worth control rod fully withdrawn and the remaining control rods fully inserted. Calculated shutdown margin is verified using startup critical data. At a minimum, this verification is performed at BOC for each reactor.

Bases. Shutdown margin is calculated on a cycle specific basis using NRC-approved methodology.[6,29] If necessary, shutdown margin is calculated at several cycle exposure points in order to determine the minimum shutdown margin for a cycle. The calculated shutdown margin is reported on a plant and cycle specific basis as required in Reference 43. SPC also confirms the worth of the standby liquid control system on a cycle specific basis using the Technical Specification values of boron concentration.

4.3 Generic Analyses

When practical, SPC performs generic analyses of normal operation anticipated operational occurrences and postulated accidents. SPC currently has two NRC-approved generic analyses, Control Rod Withdrawal Error for BWR/6[44,45] and Loss of Feedwater Heating.[46]

4.3.1 Control Rod Withdrawal Error, BWR/6

SPC had performed, and the NRC has approved, a BWR/6 generic control rod withdrawal error analysis.[44] The generic analysis has been extended to cover MEOD operation.[45] SPC demonstrated that at a 95% confidence level, that MCPR safety limits are not violated for at least 95% of the rod withdrawal events.

The SPC generic analysis of the rod withdrawal error event assumes the presence of the Technical Specification on rod withdrawal limits as a function of power which have been established by the reactor designer. These limits are one foot for core powers greater than 70% of rated power and two feet for core powers between 20% and 70% of rated. The generic analyses were performed to establish the values of operating limit MCPR as a function of core power which are required to assure that the Safety Limit MCPR will not be violated for the rod withdrawal event.

The calculational methods and procedures used to simulate the rod withdrawal event are essentially the approved methods and procedures described in Reference 29. The only difference being the different means of affecting the rod block.

4.3.2 Loss of Feedwater Heating

SPC had developed, and the NRC has approved, a generic methodology for evaluating the loss of feedwater heating (LFWH) transient in BWRs.[46] The generic methodology is a parametric description of the critical power ratio response that was developed using the results of many applications of the currently approved plant/cycle specific methodology.[6,29] SPC developed a critical power ratio function

Applying this function yields a conservative MCPR operating limit for the LFWH event.

4.4 Criticality Safety Analysis

SPC performs criticality safety analyses of new fuel storage vaults and spent fuel storage pools. Storage array k-eff calculations are performed with the KENO.Va Monte Carlo code, which is part of the SCALE 4.2 Modular Code System (Reference 50). The CASMO-3G (Version 4.1) bundle depletion code with the H library (Reference 6) is used to calculate k_∞ values for fuel assemblies at beginning of life (new fuel storage) or as a function of exposure, void, and moderator temperature for incore geometry (spent fuel storage).

The KENO.Va and the CASMO-3G computer codes are widely used throughout the nuclear industry. They are used primarily for criticality safety and core physics calculations, respectively. SPC has broad experience using both of these codes.

SPC uses criteria given in plant technical specifications and References 51 through 55 to assess the acceptability of the criticality safety analyses.

Table 4.1

Summary Description of Nuclear Design Criteria

Section	Description	Generic Design Criteria
4.2.1	Fuel Rod Power History	LHGR Limits
4.2.2	Kinetic Parameter	
	Void Reactivity Coefficient	Negative
	Doppler Reactivity Coefficient	Negative
	Power Coefficient	Negative
4.2.3	Stability	Decay Ratio <1.0
4.2.4	Core Reactivity Control	Technical Specification - Margin Maintained

5.0 EVALUATION OF ANTICIPATED OPERATIONAL OCCURRENCES

Analyses are performed to demonstrate that the fuel performs within design criteria for boiling transition during infrequent and moderately frequent anticipated operational occurrences and to establish appropriate operating limits for the reactor. The methodology used for the analysis of these anticipated events has been reported in References 3, 4, 5, 6, 29 and 40. The purpose of this section is to identify the potentially limiting events which require evaluation for each operating cycle.

To prevent or minimize boiling transition, the operating limits established by the evaluation of anticipated operational occurrences consist of a limiting transient ΔCPR, which in turn defines the MCPR operating limit, and a reduced flow MCPR limit function which adjusts the MCPR operating limit at reduced flow settings to allow for the consequences of events which are more severe at reduced flow. In some instances, these analyses also require a reduced power MCPR limit function which protects the fuel from the consequences of events which are more severe at reduced power.

5.1 Analysis of Plant Transients at Rated Conditions

Anticipated operational occurrences involving the entire core and the recirculation system are evaluated at full power and flow conditions to determine the nominal MCPR limit. Some events are also analyzed at off-rated conditions as discussed in Sections 5.2 and 5.3. The limiting transient event (or events) is (are) evaluated using the plant transient methodology described in References 4 and 5.

The evaluation of anticipated operational occurrences at rated conditions considers events in the following classifications:

- Rapid vessel pressurization;
- Decrease in recirculation flow rate;
- Increase in recirculation flow rate;

- Decrease in core inlet subcooling;
- Increase in core inlet subcooling;
- Decrease in vessel coolant inventory;
- Increase in vessel coolant inventory; and
- Combination events.

Events under the classifications of rapid vessel pressurization, increase in recirculation flow rate, and increase in core inlet subcooling are potentially limiting events. Events in the categories of increase in vessel coolant inventory and combination events are evaluated on a generic basis for each major BWR plant type. Representative analyses of potentially limiting events in the above classifications for BWR/3 plants are contained in Reference 4.

5.2 Analyses for Reduced Flow Operation

The transient events described in the preceding section are most severe at full power (except for those which result in an increase in recirculation flow and possibly those which result in an increase in vessel coolant inventory). Protection of the MCPR Fuel Cladding Integrity Safety Limit is assured during reduced flow operation through application of a flow dependent MCPR operating limit which is established independently of the full flow MCPR limits through analyses of the flow dependent transients from reduced power and flow settings.

The reduced flow MCPR limit is established to perform two protective functions. During operation in the Automatic Flow Control (AFC) mode, the limit assures that MCPR will not be below the MCPR operating limit if the flow control system demands an increase to full power and full flow. During operation in the Manual Flow Control (MFC) mode, the limit assures that MCPR will not be below the MCPR Fuel Cladding Integrity Safety Limit if the recirculation flow is inadvertently increased to the maximum allowed by the physical settings of the equipment.

Analyses are performed from various points on the power-flow operating map to demonstrate the adequacy of the flow-dependent MCPR limit to provide the desired degree of protection of the MCPR limits.

A special case of operation at less than rated power and flow is operation with a single recirculation loop out of service. It may be desirable to operate the reactor on a single loop if one component should require extensive maintenance. Analysis of single loop operation is performed on a plant specific basis, where needed.

5.3 Analysis for Reduced Power Operation

Reactor operation at points below full power is evaluated to determine the adequacy of the operating limits to protect against fuel failures during events initiated from high flow and low power conditions. If a need for reduced power operating limit augmentation is shown, results of these analyses are used to establish a power dependent MCPR limit function which protects the MCPR Fuel Cladding Integrity Safety Limit during the occurrence of anticipated events from power-flow states below the nominal 100% flow control line.

5.4 ASME Overpressurization Analysis

An overpressurization analysis is performed to assure that the vessel pressure requirements of the ASME Code are satisfied. This analysis, which presumes failure of all non-safety grade components, does not contribute to the determination of thermal margin requirements.

The turbine trip or generator load rejection transient is generally more limiting in regard to thermal margin requirements than the containment isolation event. However, the Main Steam Isolation Valve (MSIV) closure event with the assumption of failure of the direct position scram may result in a more severe calculated overpressurization. Determination of the most severe combination of overpressurization event and active component failure is accomplished generically for each major classification of BWR plants. The ASME overpressurization event is analyzed with COTRANSA2.

5.5 Limiting Transient Events

The loading of fresh fuel, regardless of design, into a reactor core may alter the characteristics of both steady state core performance and plant transient response throughout each subsequent cycle of operation. Limiting criteria for plant operations are established to assure that acceptable thermal operating margins are maintained during all anticipated operations. Application of SPC's methodology as described in Reference 4 provides a basis for the determination that plant operation will meet appropriate safety criteria.

Several of the potential events are characteristically self-limiting, but others will require reevaluation of the margins for safe operation. The following have been identified as potentially requiring reanalysis by SPC.

- turbine/generator trip w/o bypass
- loss of feedwater heating
- recirculating flow increase events
- feedwater flow increase
- inadvertent ECCS high-pressure subsystem startup
- turbine or generator trip w/o bypass and w/o direct scram*
- MSIV closure with indirect scram.*

Other events including loss of recirculating coolant flow or increase of recirculating coolant enthalpy are inherently non-limiting due to the characteristic negative void reactivity feedback of a BWR. The events described above were also found to be the most limiting of their type due to the severity of the initiating cause.

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*Evaluated for compliance with the provisions of the ASME code.

Primarily because of the strong void reactivity feedback characteristic of a boiling water reactor, incidents involving decrease in vessel coolant inventory and flow and events involving a decrease in vessel coolant subcooling are not expected to result in a limiting ΔCPR. Events involving either an increase in core inlet subcooling or rapid pressure increases are considered potentially limiting transients for all BWR designs.

A decrease in feedwater enthalpy may result in a gradual core heatup until the high neutron flux scram setpoint is exceeded. Since the gradual nature of the power excursion assures that the fuel thermal response will not greatly lag the neutronic response, this event can be evaluated with either a transient code or a steady-state code. The possible mitigation of this event with an effective flow control system would not normally be assumed in the analysis.

Rapid pressure increases may be a thermal margin limiting event for some designs and conditions. The severity of the event is strongly dependent upon the reactivity state of the core, the valve closure characteristics initiating the event, and the performance of the scram shutdown system. Thus, specific event sequences at some reactor conditions may emerge as consistently most limiting in nature. Lead plant analyses will initially evaluate a spectrum of pressurization events for each general class of JP-BWRs. Each potentially limiting event will be considered in the determination of cycle limiting conditions for operation. The scope of plant specific analyses may be narrowed to those conditions determined to be consistently more limiting.

The remaining two single event categories which involve increases in either coolant flow rate or inventory are dependent upon plant design and conditions. Both involve potentially limiting conditions at partial power and flow conditions, where the augmentation of flow (either recirculatory or feed) to the maximum physical capacity of equipment is greatest. Effective designs and/or reactor protection systems may substantially mitigate the rate and potential acceleration of power production in the core or terminate the transient prior to serious degradation of thermal margin. Current technical specifications for JP-BWRs provide an augmentation of the CPR operating limit to protect against potentially greater transient reduction of CPR at partial flow conditions. SPC will verify that the existing augmentation procedure (i.e., K_f curves) for a plant will provide adequate thermal margin for the cycle design at applicable conditions.

Once the applicable set of limiting transients for thermal margin has been identified for a specific reactor, the evaluation of each event at limiting reactor conditions will provide the basis for determining the MCPR Operating Limit which is applicable to all other anticipated operating conditions.

5.6 Control Rod Withdrawal Error

Withdrawal of the highest worth control rod in the core until its movement is blocked by the control system is evaluated with MICROBURN-B, which is described in Reference 6.

For the analysis, the reactor is assumed to be in a normal mode of operation with the control rods being withdrawn in the proper sequence and all reactor parameters within the Technical Specification limits and requirements. The most limiting case is when the reactor is operating at power with a high reactivity worth control rod fully inserted.

Results for the control rod withdrawal error analysis include maximum control rod withdrawal distance, change in thermal margin, and the limiting control rod pattern used for the analysis. For reactors utilizing reduced power augmentation to MCPR limits, the existing reduced power limit functions are revised as necessary and verified for operation with SPC-fabricated fuel. A detailed description of the SPC Control Rod Withdrawal Error evaluation methodology is given in Reference 29.

5.7 Determination of Thermal Margins

The results of the anticipated operational occurrences evaluated under this chapter are compared for the greatest change in MCPR for full power operation. (If the event causing the greatest change in MCPR is one which is sensitive to the uncertainties in plant measurement data, the statistical methods described in Reference 40 are used to calculate the thermal margin requirement. If necessary, additional statistical analyses are used to determine which of the considered transient events should define the MCPR operating limit.)

The limiting transient ΔCPR which is used to define the MCPR operating limit is used to select the rod block monitor setting from the tabulated results of the control rod withdrawal error analysis. Observance of the operating MCPR limit and rod block monitor settings determined in this fashion provides protection of the MCPR Fuel Cladding Integrity Safety Limit during operation at rated conditions.

The results of the reduced flow and reduced power analyses are used to establish the proper values for the MCPR limit functions required for operation at lower than rated power and flow conditions. Reactor operation within the power- and flow-dependent limits defined in this fashion assures adequate protection of MCPR limits throughout the power-flow operating map.

The scram insertion time used for the transient analyses may be based on either the Technical Specifications or plant measurement data. If plant measurement data were used to determine the scram performance used to define any of the limiting safety system settings or limiting conditions for operation, surveillance procedures are specified to determine the continued applicability of the data base and to modify limits to assure applicability of the analysis.

The core power and exposure distributions are monitored by the licensee throughout the cycle to assure that the EOC axial power shape assumed in the licensing analysis will bound the actual EOC axial power shape.

5.8 Supporting Documentation

Results of the transient analyses at rated conditions are documented in the plant transient analysis report, which also reports the calculation of the MCPR Fuel Cladding Integrity Safety Limit. If applicable, results from reduced flow and power analyses are also reported in the plant transient analysis report. Control rod withdrawal error analyses at reduced power conditions are performed and reported on a generic basis for the classifications of BWR plants which utilize reduced power augmentation to MCPR limits.

Results of the cycle analyses described in this section are documented in the reload analysis report, which may also incorporate the plant transient analysis report referenced above.

6.0 ANALYSIS OF POSTULATED ACCIDENTS

Hypothetical Loss of Coolant Accidents (LOCAs) are analyzed in accordance with Appendix K modeling requirements using the ECCS models described in References 21, 22, 25, and 27. Postulated Control Rod Drop Accidents are analyzed using the COTRAN methodology described in Reference 29.

6.1 Loss of Coolant Accident Analysis

The ECCS analyses provide Peak Cladding Temperature (PCT) and peak local metal-water reaction (MWR) values and are used to define MAPLHGR limits in accordance with 10 CFR 50.46. For each SPC fuel type and limiting single failure, limiting break calculations are undertaken to determine the MAPLHGR, PCT, and MWR values over the expected exposure lifetime of the fuel. The limiting break is determined generically for each BWR type by evaluating a spectrum of potential break locations and sizes. A representative fuel type is used for the break spectrum evaluation. Once the limiting break is identified, PCT analysis for specific fuel types is performed.

6.1.1 Break Location Spectrum

Representative LOCA analyses for piping breaks in the recirculation system piping form the basis for the location spectrum evaluation, which is accomplished on a generic basis for each major class of BWR plants. A figure of merit is drawn from MAPLHGR, PCT, and MWR values calculated for consistent exposure conditions in the fuel at each of the break spectrum locations. Analyses performed by the NSSS supplier are used as guidelines to narrow the scope of the analyses.

6.1.2 Break Size Spectrum

Once the location of the limiting break has been established, representative analyses are undertaken to establish the size of the limiting break. These analyses are performed on a generic basis for each major class of BWR plants.

Hypothetical piping system breaks are evaluated up to and including those with a break area equal to twice the cross-sectional area of the largest pipe in the limiting break area. Due to physical phenomena observed during the blowdown phase of the LOCA analysis, the difference in results between guillotine pipe breaks and split pipe breaks is not significant for total break areas greater than 40% of the maximum guillotine break area. Smaller breaks are evaluated to assure that the largest break is also the most severe. If the largest break is not shown to be the most severe, the most severe break is selected for further analysis. As with the location spectrum, the determination of the limiting break size is based on a comparison of MAPLHGR, PCT, and MWR values for consistent exposure conditions in the fuel.

6.1.3 Worst Single Failure

SPC works closely with the licensee to identify the most damaging single failure of ECCS equipment. The most damaging single failure of ECCS equipment is that failure which results in the largest increase in PCT for the limiting break location and size.

Previous evaluations by the NSSS vendor identifying the worst case single failure of ECCS equipment are used in determining the worst single failure. The applicability of the previous evaluations will be reviewed on a case-by-case basis. Additional calculations may be performed if plant modifications have occurred.

6.1.4 MAPLHGR Analyses

After the location and size of the limiting break have been determined, analyses are undertaken to characterize the maximum steady-state nodal power at which the fuel may be operated prior to the postulated design basis LOCA without exceeding the ECCS limits specified in 10 CFR 50.46.

The blowdown phase is evaluated with RELAX (References 22 and 27). Refill and reflood are evaluated with FLEX (References 22 and 27). Fuel heatup is analyzed with HUXY (Reference 47). Stored energy and fuel response characteristics are determined with RODEX2 (Reference 25).

6.1.5 Supporting Documentation

The break spectrum analyses are performed and reported generically for each major classification of BWR plants. The cycle specific reload analysis references the generic break spectrum analyses in specifying the limiting break location and size for the reactor in question. Detailed MAPLHGR analyses may be incorporated in the reload analysis or may be reported separately and referenced in the reload analysis.

6.2 Control Rod Drop Accident Analysis

Analysis of the postulated Control Rod Drop Accident (CRDA) is performed on a generic basis in Reference 29. Because the behavior of the fuel and core during such an event is not dependent upon system response, a single generic CRDA analysis can be applied to all BWR types. The applicability of the generic CRDA analysis is verified for each application.

The results of the generic CRDA analysis consist of deposited fuel enthalpy values parameterized as a function of effective delayed neutron fraction, Doppler coefficient, maximum (dropped) control rod worth, and four-bundle local peaking factor. Each cycle-specific application includes the values for each of the parameters, which are compared to the generic analysis and curves and the resulting deposited fuel enthalpy determined.

6.3 Fuel Loading Error

Two separate incidents are analyzed as part of the fuel misloading analysis. The first incident, which is termed the fuel mislocation error, assumes a fuel assembly is placed in the wrong core location during refueling. The second incident, the fuel misorientation error, assumes that a fuel assembly is misoriented by rotation through 90° or 180° from the correct orientation when loaded into the reactor core. For both the fuel mislocation error and the fuel misorientation error, the assumption is made that the error is not discovered during the core verification and the reactor is operated during the cycle with a fuel assembly misloaded.

The limiting parameter of interest for the fuel misloading error is the MCPR for the misloaded fuel assembly. The fuel misloading analysis determines the difference between the MCPR for the correctly loaded core and the MCPR for the core with a misloaded fuel assembly. Criteria for acceptability of the fuel misloading error analyses are that the off-site dose rate due to the event shall not exceed a small fraction of the 10 CFR 100 limits (Reference 17).

6.3.1 Misloaded Fuel Bundle

The inadvertent misloading of a fuel bundle into an incorrect core location is analyzed with the MICROBURN-B methodology described in Reference 6.

Detailed application of the misloaded fuel bundle methodology is described in Reference 29.

6.3.2 Misoriented Fuel Bundle

The inadvertent rotation of a fuel bundle away from its intended orientation is evaluated with the CASMO-3G methodology described in Reference 6. The analysis describes a minimum value of MCPR and a maximum LHGR associated with the orientation error.

Detailed application of the misoriented fuel bundle methodology is described in Reference 29.

6.4 Fuel Handling Accident During Refueling

The introduction of a new mechanical fuel design into a reactor core must be supported by an evaluation of the fuel handling accident for the new fuel design. When required, SPC performs an incremental evaluation of the impact of the new fuel design on the fuel handling accident scenario defined in the reactor's Final Safety Analysis Report (FSAR). Using the boundary conditions and conservative assumptions given in the FSAR and the relevant characteristics of the new fuel design, SPC calculates a conservative number of fuel rods expected to fail as a result of a fuel handling accident with the new fuel design.

The radiological consequences of a fuel handling accident for a new mechanical fuel design are assessed

7.0 REFERENCES

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

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54. Design Requirements for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants, ANSI/ANS American National Standard 57.2-1983, American Nuclear Society, October 1983.

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APPENDIX A

LISTING OF SIEMENS POWER CORPORATION NRC-APPROVED LICENSING TOPICAL REPORTS SUPPORTING BWR METHODOLOGY

EMF-94-217(NP)
Revision 1
Issue Date: 11/13/95

Boiling Water Reactor
Licensing Methodology Summary

Distribution

M. L. Hymas
J. H. Riddle/ComEd (10)
Document Control

Attachment F

Siemens Power Corporation Affidavit for Attachment E

AFFIDAVIT

STATE OF WASHINGTON)
) ss.
COUNTY OF BENTON)

I, R. A. Copeland being duly sworn, hereby say and depose:

1. I am Manager, Product Licensing, for Siemens Power Corporation ("SPC"), and as such I am authorized to execute this Affidavit.

2. I am familiar with SPC's detailed document control system and policies which govern the protection and control of information.

3. I am familiar with the topical report EMF-94-217(P), Revision 1, entitled "Boiling Water Reactor Licensing Methodology Summary," referred to as "Document." Information contained in this Document has been classified by SPC as proprietary in accordance with the control system and policies established by SPC for the control and protection of information.

4. The Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by SPC and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in the Document as proprietary and confidential.

5. The Document has been made available to the U.S. Nuclear Regulatory Commission in confidence, with the request that the information contained in the Document will not be disclosed or divulged.

6. The Document contains information which is vital to a competitive advantage of SPC and would be helpful to competitors of SPC when competing with SPC.

7. The information contained in the Document is considered to be proprietary by SPC because it reveals certain distinguishing aspects of SPC licensing methodology which secure competitive advantage to SPC for fuel design optimization and marketability, and includes information utilized by SPC in its business which affords SPC an opportunity to obtain a competitive advantage over its competitors who do not or may not know or use the information contained in the Document.

8. The disclosure of the proprietary information contained in the Document to a competitor would permit the competitor to reduce its expenditure of money and manpower and to improve its competitive position by giving it valuable insights into SPC licensing methodology and would result in substantial harm to the competitive position of SPC.

9. The Document contains proprietary information which is held in confidence by SPC and is not available in public sources.

10. In accordance with SPC's policies governing the protection and control of information, proprietary information contained in the Document has been made available, on a limited basis, to others outside SPC only as required and under suitable agreement providing for nondisclosure and limited use of the information.

11. SPC policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

12. Information in this Document provides insight into SPC licensing methodology developed by SPC. SPC has invested significant resources in developing the methodology as well as the strategy for this application. Assuming a competitor had available the same background data and incentives as SPC, the competitor might, at a minimum, develop the information for the same expenditure of manpower and money as SPC.

THAT the statements made hereinabove are, to the best of my knowledge, information, and belief, truthful and complete.

FURTHER AFFIANT SAYETH NOT.

SUBSCRIBED before me this 13th

day of November, 1995.

Susan K. McCoy
NOTARY PUBLIC, STATE OF WASHINGTON
MY COMMISSION EXPIRES: 1/10/96