WESTINGHOUSE CLASS 3

WCAP-10263

A REVIEW PLAN FOR UPRATING THE LICENSED POWER OF A PRESSURIZED WATER REACTOR POWER PLANT

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ABSTRACT

This report defines a review plan for increasing the licensed power rating of a nuclear plant. It describes the evaluations required to support an uprating application for a typical plant, and proposes a basis for setting the ground rules and criteria for performing those evaluations. Its purpose is to develop guidelines for licensees to use when applying for increases in their licensed power ratings.

The review plan is based on three propositions fundamental to the feasibility of uprating an operating nuclear power plant:

1. Power related aspects of the plant design will be reviewed.

2. The licensing criteria and acceptance standards applicable to current plant operation will apply to uprated plant operation.

3. Analyses required to support an uprating application will be performed using current analytical techniques.

The NRC must establish its position regarding these issues in order for the applicant to provide sufficient and appropriate information in support of an uprating application.

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1.0 INTRODUCTION

1.1 BACKGROUND

Due to the increasing lead time and the rising capital cost of new power plant construction, there has been a major trend by electric utilities to upgrade and uprate their existing generating plants. Increasing the number of kilowatt hours generated by an existing unit is a cost effective way to add generating capacity that benefits both the utility and its customers. Most of the uprating effort to date has been concentrated on fossil fueled plants. Although there is a growing interest in uprating nuclear plants as well, many utilities have hesitated to pursue that option because the regulatory review and approval process is not clear at present. The impact of an uprating application on the current operating license, the criteria that will be applied by regulatory ..orities in their review of an uprating application, and the time required to complete the review process are all critical factors in determining if it is feasible to uprate a nuclear plant.

1.2 HISTORY

Thermal power uprating of nuclear facilities is not a new concept. During the 1960's and early 1970's a number of utilities and NSSS suppliers recognized the potential for uprating the thermal output of the nuclear unit to increase electrical generation. Conservatism was designed into the original plant systems and equipment with the understanding that increased thermal power ratings would be requested at a later date based on the levels of safety and operability demonstrated by the plant at the originally licensed power. The Robert E. Ginna and H. B. Robinson II nuclear units are examples of Westinghouse plants uprated after the initial operating license was granted. Ginna was operated at a rating of 1320 MWt until an amendment to increase the licensed rating to 1520 MWt was approved. H. B. Robinson II was originally operated at 2200 MWt until a thermal power uprating to 2300 MWt was approved. Later plants have been uprated before initial power generation. The D. C.

Cook Unit II, for example, was uprated from 3250 MWt to 3403 MWt during licensing of the plant. Several non-Westinghouse nuclear facilities have also been uprated, including Fort Calhoun, St. Lucie I, Crystal River, and Millstone II. Today there is a broad base of experience to support the operation of plant components at uprated levels. In an effort to streamline and standardize the licensing review process, nuclear suppliers have standardized plant, component and system designs to envelope a spectrum of operating conditions over a broad range of thermal power ratings. Tables 1 through 3 show the progression of Westinghouse NSSS ratings with time for 2, 3 and 4 loop plants with an active fuel length of 12 feet. From the tables it can be seen that over the years thermal power has increased by approximately 30 percent. During this period, many of the standard NSSS components have been licensed and operated at power levels beyond those of their initial application.

It is also significant that the safety related features of a Westinghouse PWR are typically designed for a thermal power rating about five percent greater than the licensed rating. This power rating is referred to as the Engineered Safeguards Design Rating (ESDR), and it is usually determined by the turbine limiting flow capability. As a result of this practice, many of the Westinghouse pressurized water reactors operating today could be uprated to the ESDR with only minor software and hardware modifications. With appropriate modifications to the NSSS and to the BOP, some of these units could be uprated beyond the ESDR.

1.3 OBJECTIVES

The primary objective of this report is to develop guidelines for licensees to use when preparing applications for increases in their licensed power levels. It consists of two principal elements. One describes the safety evaluations and component design reviews that will be performed to demonstrate that a plant can continue to be operated without undue risk to the health and safety of the public if the licensed power level is increased as requested. The other proposes a set of ground rules and criteria that provide a uniform and well-defined base from which to evaluate changes in power rating. It is hoped that, through review of this report and discussions that follow, the NRC will establish:

1) A position regarding the information required to permit the staff to conclude its review of an uprating application; and

2) A basis for defining the ground rules and criteria that will be used in evaluating that application.

2.0 UPRATING REVIEW PROCESS

2.1 GROUND RULES AND CRITERIA

In order to prepare an uprating application for submittal to the NRC, a licensee must be able to establish:

1. The potential impact of the uprating application on the current design basis

2. Scope of regulatory authority review

3. Applicable regulatory codes, standards and criteria

4. Analytical techniques to be utilized

The NRC position regarding these issues will have a major impact on the feasibility of uprating nuclear facilities. It will also facilitate the review process if the applicant is able to provide sufficient and appropriate information to support the initial uprating application. Following is a discussion of the Westinghouse position on these issues.

2.1.1 IMPACT ON CURRENT DESIGN BASIS

The proposed uprating will be analyzed in accordance with the codes and standards applicable to the plant at the time of submittal and, as such, will have no impact on the plant design basis.

2.1.2 SCOPE OF REVIEW

The scope of regulatory review should encompass all aspects of the facility design and operation which are impacted by the proposed uprating. Any aspect of the design that is impacted will be evaluated against the current codes and regulations applicable to the plant. However, a review will be made as defined in 10CFR50.59 to identify any potential unreviewed safety questions that might result from the uprating. Section 3 of this report provides a discussion of the scope of a typical uprating review.

2.1.3 CODES, STANDARDS, CRITERIA

The proposed uprating will be performed in accordance with the established licensing criteria and standards which apply to the current operating license of the specific plant being uprated. If the uprating involves a potentially unreviewed safety question, it will be identified and resolved during the uprating review process. This process will assure that protection of the public health and safety can be maintained within the current licensing basis.

The need for plant modifications associated with the uprating will be established by the results of component design reviews and analytical evaluations based on operating conditions at the uprated power. These reviews and evaluations will be used to identify any areas where existing plant components and designs fail to meet applicable licensing criteria and standards at the uprated power, as well as to determine appropriate modifications to re-establish compliance. The types of modifications which might be required to support a plant uprating are judged not to be "material alterations" under 10CFR50.91 because they would not change the plant operations or purpose as originally licensed.

2.1.4 ANALYTICAL TECHNIQUES

The technology and data base of the nuclear industry have progressed significantly in many areas. To take advantage of that progress, current analytical techniques will be used for any analyses required to support an uprating. This will also facilitate performance of the analyses and the regulatory review of the results. Existing analyses will not be redone if they are not affected by the uprating, or if they have already been analyzed at the uprated power for the FSAR.

2.2 UPRATING REVIEW PROCESS

Table 4 summarizes the major milestones that must be accomplished during review and approval of a typical uprating. These milestones are applicable to upratings in general, and can be modified easily to suit the specific requirements of a particular uprating application. A discussion of the more significant interface activities in the uprating process follows.

2.2.1 UPRATING PARAMETERS

The initial step in an uprating program is for the utility to establish uprating parameters and to define an associated plant configuration for the evaluation of limiting plant transients and accidents. This evaluation is performed to confirm that compliance with the established plant licensing basis will be maintained with the proposed uprated parameters and plant configuration. Based on the results of this evaluation, the utility determines the feasibility of proceeding with the uprating program.

2.2.2 PRE-TENDERING DISCUSSIONS

The utility will initiate pre-tendering discussions to inform the NRC of the impending uprating application, and to describe the proposed uprating program. This will permit the commission to plan and schedule the uprating review, and to provide comment on the utility uprating program. It is assumed that the NRC will have previously provided guidance on the program content through its comment on this report. Based on these pre-tendering discussions, the utility decides whether or not to make a final commitment to the uprating program.

The utility, NSSS supplier, and architect engineer will then meet with the NRC in a technical review of the evaluation and analysis of limiting transients and accidents. Results of this discussion are documented to the NRC for information, informal review and schedule planning. Following this meeting, the NRC responds with a schedular commitment, and identifies any technical constraints that could inhibit licensing of the uprated conditions.

2.2.3 DOCKETING AND APPROVAL

Based on comments from the NRC, the remainder of the uprating program is executed (e.g., evaluations, analyses and hardware modifications). A final licensing document is submitted containing all required analyses and evaluations and describing any required plant modifications to demonstrate that compliance with the established licensing criteria is maintained. This document is docketed, and forms the basis for final NRC review and approval of the uprating.

2.2.4 UPRATING IMPLEMENTATION

After the NRC has issued a license amendment for the uprated conditions, the utility implements the uprating. Plant design and operating documents are revised consistent with parameters for the uprated power. Hardware modifications are completed and verified functional. When these actions are complete, the plant can be operated at the uprated power. The next periodic updating of the plant Final Safety Analysis Report required by 10CFR50.71 will incorporate changes resulting from the plant uprating.

3.0 SCOPE OF REVIEW FOR A PLANT UPRATING

3.1 GENERAL

The licensing review for a plant uprating is typically performed in two parts. In the first part, design limiting conditions and events are reviewed to demonstrate the feasibility of uprating to the desired power. This information is also used as a basis for pre-tendering discussions in which feedback from the NRC is obtained to identify any licensing constraints. The review is then completed by performing all of the remaining evaluations and analyses required to license the uprating.

3.1.1 DESIGN LIMITING UPRATING EVALUATIONS

Initially, a set of plant parameters will be established as a basis for the uprating evaluations. These parameters will be established by the utility in conjunction with the NSSS supplier and architect engineer based on a knowledge of replicate plants/systems operating at higher power levels, available system/component margin, potential hardware/system improvements available and limitations of components and systems which would not be practical to replace or modify (e.g., containment or reactor vessel structures). Key parameters include:

NSSS Power	Feedwater Flow Rate
Reactor Power	Steam Generator Outlet Pressure
Core Flow Rate	Reactor Vessel Inlet Temperature
Reactor Coolant Pump Flow Rate	Reactor Vessel Outlet Temperature
Steam Flow Rate	Steam Generator Feedwater Temperature

As the program progresses, these parameters will be used to determine more detailed plant parameters, such as heat rejection rates to the component cooling water systems, mass and energy release rates, radiation source terms and emergency core cooling system parameters.

Evaluation of the design limiting accidents and transients are performed next to determine the adequacy of the existing plant for operation at uprated conditions. These evaluations will also provide input to define any plant modifications that might be required to satisfy the acceptance criteria. All analyses will be made to FSAR quality standards using NRC approved calculational techniques so that they need not be re-done during the balance of the uprating evaluations.

Accidents and transients that would be analyzed during this part of a typical plant uprating review include design limiting events for

- DNB Margin
- Reactivity Excursions
- ECCS Capability
- Peak RCS Pressure
- Heatup
- Auxiliary Feedwater System
- Containment Design

In parallel with the review of the design limiting accidents and transients, an analysis of the NSSS systems and components will be performed to determine their capability for operation at the uprated power. These analyses and evaluations will either 1) verify compliance of existing systems and operating procedures with applicable plant design bases and regulatory requirements, or 2) identify those areas where revisions and/or modifications are required. This review will include all of the classical NSSS fluid systems components listed in Table 5, as well as any components provided by the NSSS supplier in optional systems. The impact of the uprated parameters on functional design requirements and structural integrity of these components will be reviewed. Typical NSSS operating transients to be considered during this review are listed in Table 6. Where the uprating requirements are not bounded by current component design, revisions and modifications will be made as necessary to demonstrate compliance with applicable codes and standards.

The plant technical specifications will be reviewed to identify required revisions to protection setpoints and/or limiting conditions for operation.

3.2 DETAILED NSSS EVALUATION

3.2.1 GENERAL

The detailed evaluation will differ from the design limiting evaluation in that it is focused on those specific areas in which the need for further evaluation and possible plant changes has been identified.

When the design limiting evaluation has indicated that the uprating has an impact on a particular system and/or component, the designer will receive revisions to the design bases and/or functional requirements for the specific system/component and will determine if the installed system/component remains in compliance with the plant specific standards, design criteria, and regulatory requirements for the uprated conditions.

An uprating can increase the operating power level and temperatures of the RCS. This necessitates the verification that each installed system component and the associated analyses are in compliance with the design codes, standards and criteria for the revised nominal operating conditions. In some instances it will be necessary to revise the documented analyses to account for the increased power level. Three levels of effort may be necessary to accomplish this review. Each of the three levels is discussed below:

The first level of effort is to identify for which NSSS systems and associated components no change in the original design bases and functional requirements is required. For these components and/or systems, no additional effort is required with respect to the uprating.

The second level of effort is to identify for which NSSS components the uprated conditions are bounded by analyses performed for a generic design or for a plant with the identical systems component at power levels equal to or greater than those associated with the proposed change. For these cases, an evaluation is provided to document the acceptability of the installed system or component.

The third level of effort is to confirm compliance with the applicable design codes, standards and criteria for specific instances where the uprated conditions are not bounded by analyses performed for a generic design or for a unit with the identical components at duty ratings equal to or greater than those associated with the proposed change.

In summary, the majority of the NSSS components will be enveloped by either the original analyses for the specific unit or analyses for other plants with identical structures at a higher duty rating. For specific components where additional analyses are necessary, it must be determined if the structures remain in compliance with the design codes, standards and criteria applied to the current license for the specific unit. Should it be necessary, appropriate action will be taken to assure compliance with the unit's current licensing bases at the uprated condition.

3.2.2 REACTOR COOLANT SYSTEM (RCS)

As a minimum, the impact of the proposed uprating on the functional, operational, and safety related aspects of the RCS will be evaluated and/or analyzed in the following areas:

Analyses will be performed to determine the pressurizer spray, power operated relief and safety valve relief capacity necessary to maintain the original design bases for the increased power level. The specific plant Safety Analysis Report discusses the design bases for that unit. Evaluations will be performed to determine the necessary operating range of the Reactor Coolant System control, protection and measurement instrumentation (e.g., pressure, temperature, flow, level, flux mapping and nuclear power) and the associated systems (e.g., nuclear instrumentation, flux mapping, bottom mounted instrumentation and incore thermocouple systems) at the increased power level. Any necessary revisions to the current operating ranges or functional requirements will be identified.

3.2.3 CHEMICAL AND VOLUME CONTROL SYSTEM (CVCS)

All functional requirements of the CVCS will be reviewed. The areas which are most likely to be impacted by the uprating are:

1. CVCS heat exchanger heat rejection rates - If the uprating results in an increased RCS cold leg temperature, the heat loads from the CVCS heat exchangers to the component cooling water system will increase.

2. Components and systems located upstream of the letdown heat exchanger - Should the RCS cold leg temperature be increased at the uprated conditions, the uprated functional requirements may not be enveloped by the current component design bases. The capability of the components to perform at the uprated conditions will be confirmed and appropriate modifications made. Should the RCS cold leg temperature be reduced, the existing design bases would bound the uprated condition.

3.2.4 RESIDUAL HEAT REMOVAL SYSTEM (RHRS)

A higher power level results in an increase in the amount of decay heat being generated in the core during normal cooldown, refueling operations and accident conditions. This will result in a higher heat load on the residual heat exchangers during the cooldown and also during the refueling outage. The increased heat loads will be transferred to the Component Cooling Water System (CCWS) and ultimately to the Service Water Cooling System (SWCS). It will be necessary to evaluate the performance of the RHRS, CCWS and SWCS with the increased heat loads.

On some plants the RHRS pumps and heat exchangers are an integral part of the Safety Injection System (SIS). For these plants, the ability of the RHRS to meet the design and functional requirements of the SIS at the uprated conditions will be confirmed.

The uprating does not impact the ability of the RHR pumps to transfer water to or from the refueling water storage tank.

3.2.5 SAFETY INJECTION SYSTEM

The required volume, duration and heat rejection capability of the safety injection flow in the event of a break is determined based on analytical and empirical models which simulate reactor conditions subsequent to the postulated RCS and steam system breaks. As a result of these analyses the system and component requirements necessary to demonstrate compliance with regulatory requirements at the uprated power level will be established. Should the requirements fall outside the bounds of the installed system, it may be necessary to implement software/hardware modifications, provide revised heat rejection rate data for the CCWS and revise the electrical loading of the SIS equipment on the safeguards electrical systems. In the event the current SIS provides adequate safety margin, no additional effort would be required.

3.2.6 BORON THERMAL REGENERATION SYSTEM (BTRS)

Evaluations at the uprated conditions will be performed to assure that the installed system/component design bases and functional requirements bound the proposed operating conditions.

3.3 BALANCE OF PLANT SYSTEMS AND EQUIPMENT EVALUATIONS

Uprating the electrical generation capability of the unit will also have an impact on the BOP systems and equipment. As part of the evaluation of the NSSS, the NSSS/BOP interfaces will be reviewed and changes to the interface information will be provided to the utility. The review and analyses for the BOP will follow a pattern similar to the NSSS procedure as discussed in section 3.2.

Initially the plant conditions and configuration associated with the target uprating and a delineation of the necessary interface data will be identified. A review and analysis of the limiting BOP accidents (e.g. containment pressure and temperature) will be performed to confirm that the proposed uprating parameters and associated plant configuration are in compliance with the plant license.

Subsequent to the evaluation of the limiting BOP accidents, detailed evaluations of the BOP systems and equipment will be performed. If the uprated system conditions are bounded by existing documentation, no additional effort will be required for that system or the equipment in that system. If the uprated conditions are not bounded by the current design bases and functional requirements, necessary software/hardware revisions will be identified. Where revisions are identified, further evaluation will be performed to determine if the equipment remains in compliance with the plant's current licensing basis. If necessary modifications to the equipment will be identified to assure compliance with the licensing basis is maintained.

3.3.1 TYPICAL BOP/NSSS INTERFACES

The following BOP/NSSS interfaces may be impacted by the uprating. These interfaces would only be affected as a result of modifying the design bases and/or functional requirements of another NSSS or BOP system serviced by these BOP areas.

a. AC and DC Emergency Power Systems - The plant is equipped with both onsite (AC and DC) and offsite (AC) emergency electrical power systems to provide reliable power to the NSSS and BOP safety systems. Increases in the electrical power requirements of the NSSS essential systems, which result from the uprating will be identified.

b. AC and DC Power Systems - The plant is equipped with electrical power systems which supply the NSSS equipment. Any increased NSSS electrical loads which may be required as a result of the uprating will be identified.

c. Demineralized Water Makeup System - The purity requirements of the makeup water could be affected . the uprating.

d. Auxiliary Feedwater System - The Auxiliary Feedwater System supplies feedwater to the secondary side of the steam generators whenever the main feedwater system is not available, in order to maintain the steam generator as the principal reactor shutdown heat sink. This system may also function as an alternate to the Main Feedwater System during startup, hot standby and cooldown conditions. The Auxiliary Feedwater System provides core cooling during abnormal transients.

e. Mass and Energy Release to the Containment - The mass/energy release data will be employed to determine the containment pressure and temperature environment during the postulated accidents and to determine the associated loadings on the structures and components within the containment in accordance with the licensing basis of the specific unit. Mass and energy release data for the uprated conditions will be provided.

f. Spent Fuel Pit Cooling System - The functions of this system are:

   1. Maintain desired water temperature in the spent fuel pit.

   2. Maintain chemistry and activity level requirements in spent fuel pit water.

   3. Provide refueling water cleanup and purification capabilities.

The increased decay heat rates will be identified to allow evaluation of the ability of the installed spent fuel pit cooling system to maintain acceptable temperatures within the spent fuel pit. There are no NSSS/BOP interface changes with respect to the other two functions.

g. Main Steam System - The primary purpose of the steam system is to contain and transport steam from the NSSS steam generators to the main turbine. The steam system also forms part of the boundary between the radioactive fluid systems and the environment.

The uprating will result in increased steam flow and/or pressure in the main steam system.

h. Component Cooling Water System (CCWS) - The CCWS is an intermediate system between the Reactor Coolant System and the Service Water Cooling Systems (SWCS). It ensures that leakage of radioactivity from the components being cooled is contained within the plant. The system typically removes heat from the NSSS and some BOP components. Revised heat rejection rates and/or cooling water flow requirements will be identified.

i Radiological Source Terms - Radiological Source Terms are used in assessing the radiological consequences of accidents. Any changes identified as a result of uprated parameters will be identified.

k. Plant Testing - Numerous qualification and performance tests were completed for the initial startup to assure that all systems/components of the BOP and NSSS are in compliance with the design and licensing bases for the unit. These tests also establish the operating margins of the plant systems. It will be necessary to verify that the performance of any system/component modifications are in compliance with the requirements of the uprating and the licensing bases. The recommended test program for NSSS and interfacing BOP systems would be developed on a plant specific basis, depending upon the magnitude of hardware modifications and the magnitude of the uprating.

3.4 ACCIDENT ANALYSES

A reference analysis is normally established as part of the initial licensing effort as documented in the FSAR. This is supplemented by reanalyses required for reload fuel or plant equipment or system changes. For a plant uprating, a safety evaluation is performed to confirm the validity of applicable reference analyses. If the reference analyses do not bound the uprated conditions, reanalysis using currently approved methods and appropriate input parameters will be performed. The Westinghouse Reload Safety Evaluation Methodology (RSEM) report (Ref. 1) summarizes the overall process to assess changes. This report was written primarily for reloads, but the process described is also applicable to upratings.

The uprating evaluation process includes:

1. A systematic evaluation to determine a) what parameters utilized in the reference safety evaluation are impacted by a change in plant rating and b) if these new parameters are bounded by the current reference safety evaluation.

2. A determination of the effects on the reference safety analysis when a parameter per 1.b above is not bounded. This determination may require a reanalysis as appropriate.

The specific steps in this process are the design initialization, design process and safety evaluation.

The design initialization process involves the collection and review of design basis information to ensure that the uprating safety evaluation will be based on the actual fuel and core components in the plant, the actual plant operating history, and any plant system changes associated with the uprating. The review includes the utility requirements, core design parameters, safety criteria and related constraints, specific operating limitations and past operating history. The initialization review identifies the objectives, requirements and constraints for the uprated cycle being designed.

The design process ensures that the utility power and energy requirements established in the design initialization phase are achieved. The key safety parameters for the cycle (i.e. uprating and reload parameters) are then determined based on the preliminary design. The safety bases to be met for the uprated core are:

Departure from Nucleate Boiling Design Basis - There will be at least a 95 percent probability that departure from nucleate boiling (DNB) will not occur on the limiting fuel rods during normal operation, operational transients, or during any transient conditions arising from faults of moderate frequency (Condition I and II events), at a 95 percent confidence level. In order to meet this basis, the minimum allowable DNB ratio is determined. This minimum allowable DNBR depends upon the DNB correlation employed in the analysis. For example, this minimum DNBR was conservatively set at 1.30 for the original W-3 DNB correlation and 1.17 for the WRB-1 DNB correlation.

Fuel Temperature Design Basis - During modes of operation associated with Condition I and Condition II events, there is at least a 95 percent probability that the peak kw/ft fuel rods will not exceed the UO_2 melting temperature, at the 95 percent confidence level. The melting temperature of UO_2 is taken as 5080°F, unirradiated and decreasing 58°F per 10,000 MWD/MTU. By precluding UO_2 melting, the fuel geometry is preserved and possible adverse effects of molten UO_2 on the cladding are eliminated. To preclude center melting and to provide a basis for overpower protection system setpoints a calculated centerline fuel temperature of 4700°F has conservatively been selected as the overpower limit.

Reactor Coolant System Pressure - Peak RCS pressure is not to exceed 110 percent of the design pressure during Condition I and Condition II events.

Loss of Coolant Design Bases (10CFR50.46) - The LOCA design bases incorporates a review of peak cladding temperature, maximum cladding oxidation, maximum hydrogen generation, coolable geometry and long-term cooling.

Compliance with these bases ensures that the margin of safety as defined in the basis of the technical specification has not been reduced (a 10CFR50.59 requirement). These design bases are interpreted as safety limits for the safety evaluation.

The objective of the uprating safety evaluation is to verify compliance with the currently established safety limits for the specific unit with the uprated core and plant system design. This is accomplished by examining each accident presented in the FSAR or subsequent submittals to the NRC to determine if the reference analysis remains valid for the uprating. A typical listing of postulated accidents is presented in Table 7. The specific transients for each plant can be found in the unit's Safety Analysis Report. For those accidents which are affected by the uprating, an evaluation is performed to verify compliance with the applicable safety limits.

In the performance of an uprating safety evaluation, each accident is examined and the bounding values of the key safety parameters which could be affected by the uprating are determined based on the reference analysis. These parameters form the basis for determining whether the reference safety analysis remains valid. For an uprating, values of these safety parameters are determined for the core during the nuclear, thermal and hydraulic, and fuel rod design process. Each of these parameters is compared with the reference analysis value to determine if any parameter is not bounded. If all of the parameters are bounded, the reference analysis remains valid and no new analysis is needed to verify that the safety limits are not exceeded. Should one or more of the safety parameters not be bounded, a re-evaluation of the accident is performed.

The re-evaluation may be of two types. If the parameter is only slightly out of bounds, or if the transient is relatively insensitive to that parameter, a simple quantitative evaluation may be made. Alternatively, should the deviation be large or be expected to have a more significant or not easily quantifiable effect on the accident, a re-analysis of the accident is performed. If the accident is re-analyzed, the analysis methods follow standard procedures and will typically employ analytical methods which have been used in previous submittals to the NRC. These methods are those which have been presented in the FSAR or subsequent submittals to the NRC for that plant, reference SARs such as RESAR, or reports submitted for NRC approval. The re-analyzed accident must continue to meet the appropriate safety limit for that event in order to be considered to have acceptable results.

Accident re-analysis may also be necessary if there are any changes made to the reactor plant systems, either in configuration, performance or setpoints as determined during the design initialization phase. Should any plant or system changes affecting safety be incorporated, their impact will be determined during the evaluation.

Measurements of nuclear and safety related parameters during and after cycle startup serve two purposes. The first is to insure that the measured parameters fall within the limiting values included in the Technical Specifications of the plant. The second is to confirm the validity of the corresponding design calculations. For an uprating, as for any other reload, startup physics program will be performed to confirm the key safety parameters such as rod worths and moderator temperature coefficients. The testing will also confirm that the core is properly loaded. The values of all measured parameters are compared to those calculated using the design codes.

4.0 REFERENCES

1. Bordelon, F. M. et. al., WCAP-9272 Westinghouse Reload Safety Evaluation Methodology

TABLE 1

COMPARISON OF TYPICAL 2 LOOP PLANT PARAMETERS

	First Generation	Second Generation	Third Generation	Future Generation
NSSS Power, MWt	1520	1650	1882	1967
NSSS System Pressure Nominal, psia	2250	2250	2250	2250
Total Core Inlet Thermal Flow Rate, gpm	179,400	178,000	189,000	189,000
Reactor Coolant System Temperature, °F				
Nominal Reactor Vessel/Core Inlet	552.5	535.5	549.9	553.0
Average Rise in Vessel	57.3	63.6	66.2	68.6
Average in Vessel	581.2	567.3	583.0	587.5
No Load	547	547	557	557
Rated Steam Pressure, psia	821	750	920	920
Major Components				
Fuel Type	14 x 14	14 x 14	16 x 16	16 x 16
Steam Generator Model	44	51	F	F
Reactor Coolant Pump Model/Horsepower	93/6000	93A/6000	93A/7000	93A/7000

TABLE 2

COMPARISON OF TYPICAL 3 LOOP PLANT PARAMETERS

	First Generation	Second Generation	Third Generation	Future Generation
NSSS Power, MWt	2208	2441	2785	2910
NSSS System Pressure Nominal, psia	2250	2250	2250	2250
Total Core Inlet Thermal Flow Rate, gpm	268,500	265,500	292,800	278,400
Reactor Coolant System Temperature, °F				
Nominal Reactor Vessel/Core Inlet	546.2	543.0	557.0	552.3
Average Rise in Vessel	56.1	62.6	62.9	68.9
Average in Vessel	574.2	574.3	588.5	586.8
No Load	547	547	557	547
Rated Steam Pressure, psia	785	785	964	850
Major Components				
Fuel Type	15 x 15	15 x 15	17 x 17	17 x 17
Steam Generator Model	44	51	F	F
Reactor Coolant Pump Model/Horsepower	93/6000	93A/6000	93A/7000	93A/7000

TABLE 3

COMPARISON OF TYPICAL 4 LOOP PLANT PARAMETERS

	First Generation	Second Generation	Third Generation	Future Generation
NSSS Power, MWt	2758	3250	3423	3600
NSSS System Pressure Nominal, psia	2250	2250	2250	2250
Total Core Inlet Thermal Flow Rate, gpm	358,800	354,000	354,000	360,000
Reactor Coolant System Temperature, °F				
Nominal Reactor Vessel/Core Inlet	543.0	536.3	552.5	547.6
Average Rise in Vessel	53.0	63.0	64.3	66.7
Average in Vessel	569.5	567.8	584.7	580.9
No Load	547	547	557	547
Rated Steam Pressure, psia	776	758	910	835
Major Components				
Fuel Type	15 x 15	15 x 15	17 x 17	17 x 17
Steam Generator Model	44	51	F	F
Reactor Coolant Pump Model/Horsepower	93/6000	93A/6000	93A/6000	93A/6000

TABLE 4

TYPICAL UPRATING MILESTONES

Milestone	Est. Time (months)	Action
1. Select Target Parameters and Plant Configuration	1-2	Utility, A/E and W
2. Perform Limiting Accident Analyses	4-6	Utility, A/E and W
3. Inform NRC of Intent to Submit Uprating Application		Utility
4. Prepare and Submit Document Summarizing Limiting Accident Analyses and Identifying Scope of Implemenation Program	1-2	Utility, A/E and W
5. Review and Comment on Uprating Program	3-6	NRC
6. Perform Remainder of Uprating Evaluations and Implement Hardware Improvements:		Utility, A/E and W
Analyses	6-9	
Hardware	6-24	
7. Final Review and Approval of Uprating Program	3-6	NRC
8. Issue Operating License Amendment		NRC

TABLE 5

NSSS COMPONENTS

Reactor Vessel
Reactor Internals
Control Rod Drive Mechanisms
Incore Instrumentation Tubing
Reactor Coolant Loop Piping
Reactor Coolant Loop Isolation Valves
Pressurizer
Steam Generator
Reactor Coolant Pumps
Component and Piping Supports
Tanks
Heat Exchangers
Pumps
Valves
Filters
Evaporators
Instrumentation
Refueling and Fuel Handling Equipment
Chillers

TABLE 6

SUMMARY OF TYPICAL REACTOR COOLANT SYSTEM DESIGN
ACCIDENTS AND TRANSIENTS

Normal Conditions

1. Heatup and Cooldown at 100°F/hr (pressurizer cooldown 200°F/hr)
2. Unit Loading and Unloading at 5 percent of full power/min
3. Step Load Increase and Decrease of 10 Percent of Full Power
4. Large Step Load Decrease
5. Steady State Fluctuations

Upset Conditions

1. Loss of Load, without immediate turbine or reactor trip
2. Loss of Power (blackout with natural circulation in the RCS)
3. Loss of Flow (partial loss of flow one pump only)
4. Reactor Trip from Full Power
5. Operational Basis Earthquake (20 earthquakes of 20 cycles each)

Faulted Conditions

1. Main Reactor Coolant Pipe Break
2. Steam Pipe Break
3. Steam Generator Tube Rupture
4. Design Basis Earthquake

Test Conditions

1. Turbine Roll Test
2. Hydrostatic Test Conditions
   a. Primary Side
   b. Secondary Side
   c. Primary Side Leak Test

TABLE 7

LIST OF TYPICAL ACCIDENT ANALYSES

Uncontrolled RCC Assembly Withdrawal

1. From a subcritical condition
2. At power

RCC Assembly Misalignment

Chemical Volume and Control System Malfunction

1. Dilution during refueling
2. Dilution during startup
3. Dilution at power

Loss of Reactor Coolant Flow

1. Flow coast-down
2. Locked rotor accident

Start-up of an Inactive Reactor Coolant Loop

Loss of External Electrical Load

Loss of Normal Feedwater

Excessive Heat Removal Due to Feedwater System Malfunction

Excessive Load Increase Incident

Loss of all A.C. Power to Station Auxiliaries

Steam Generator Tube Rupture

Rupture of a Steam Pipe

Rupture of a Control Rod Drive Mechanism Housing

Reactor Coolant System Pipe Rupture