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NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

Dec. 10, 1999

MEMORANDUM TO: ACRS Members and Staff

MEMORANDUM #: [redacted]

FROM: A. W. Cronenberg

SUBJECT: FY99 Efforts on Power Uprate: Final Report and ANS Summary

Attached are a Final Report and ANS-Summary, which document my FY-99 efforts related to *A Review of Power Uprate Applications and Potential Synergistic Safety Issues*. The attached report includes revisions per internal ACRS review/comments. The report has been forwarded to cognizant NRR staff, for their input. I anticipate writing an 8-10 page conference, upon receipt of NRR comments, which will constitute closure of my efforts on the subject.

The enclosed documents are provided in advance of the ACRS Retreat, where a 20-minute presentation is to be given on the subject.

REVIEW OF POWER UPRATE APPLICATIONS and POTENTIAL SYNERGISTIC SAFETY ISSUES

Report: AWC-101.99

by

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ABSTRACT

The ACRS has reviewed, and the NRC has approved, recent power uprate applications for the Monticello and Hatch plants. Both the ACRS and agency staff have likewise reviewed the General Electric Generic Power Uprate program for boiling water reactors. These activities are in addition to several dozen uprate approvals made during the 1980s and early 1990s. Although the NRC staff has reviewed all uprate applications to assure that current regulatory requirements are satisfied, the concern exists that potential synergistic processes may occur which may not be adequately considered in such reviews. Specifically the concern is that high core power densities, when combined with system/component degradation via plant aging and fuel life extensions to high burnup, may impact safety margins. This investigation centers on an assessment of agency uprate review practices and evaluates the need to consider potential synergistic effects.

For the uprate applications reviewed in this study, namely that for the Brunswick, Hatch, Limerick, Maine Yankee, Monticello, North Anna, Surry, and Wolf Creek plants, little documentation was found with regards to consideration of potential synergistic consequences of higher core power densities when combined with component aging and high-burnup fuel effects. A review of operational events and incidents for uprated plants shows some evidence, albeit an indirect linkage, that synergistic effects may indeed occur. It is recommended that agency efforts be expedited for formulation of a Standard Review Plan (SRP) for power uprates. Review requirements and acceptance criteria should be established to address safety implications of higher power densities in combination with extended fuel burnup and component aging effects.

DISCLAIMER

The views expressed in this report are solely those of the author. They do not necessarily reflect the opinions held individually or jointly by ACRS members or the Commission.

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1. INTRODUCTION

A utility application for a reactor operating license will specify the maximum thermal power rating for the plant. This power limit serves as the basis for extensive analysis to demonstrate the operational safety of the plant, which is documented in the applicant's Final Safety Analysis Report (FSAR). In addition to the FSAR, the applicant will submit a pant-specific Technical Specification (TS) report, which specifies all plant operating procedures and technical specifications for plant systems, components, and structures. The NRC then performs an extensive review of the applicant's FSAR and TS, and documents it's findings in an agency report known as the Safety Evaluation Report (SER). The approved FSAR and TS reports form an integral part of a utilities operating license.

After a period of demonstrated operational safety at the power level specified in the Final Safety Analysis Report (FSAR), a utility will often request an amendment to its license to allow operation at a somewhat higher power. In some cases, specifically for GE-BWR plants, the request is sometimes referred as a "stretch power" increase, if the power increase is 5-% or less, because the nuclear steam supply system was designed for this higher power level. In most cases the application refers to an "extended power " or "uprated power" increase, where some of the design criteria used in the original FSAR may be limited to the original power rating. In either case, the utility will submit new supporting safety analyses at the elevated power. This re-analysis presents similar information as found in the original FSAR, with emphasis on safety considerations that might be affected by the increased power; examples being a re-evaluation of core cooling and core thermal-hydraulic conditions, DNB (departure from nucleate boiling) margins, the capacities of the residual heat removal system, emergency coolant injection, and feedwater supply. An extensive re-evaluation of design basis accidents (DBA) and off-normal transients at the higher power level must likewise be submitted by the applicant. The uprate application and associated supporting analysis are then reviewed by the NRC staff, where findings are reported in the uprate Safety Evaluation Report (SER). The scope of the NRC review also encompasses an evaluation of any potential unreviewed safety questions that might occur as a result of the increased power rating in accordance with 10CFR-Part 50.59. Changes to plant technical specification for operation at the increased power level must also be approved by the staff during its review. The uprate application is approved if the case has been made that the plant can be operated at safely at the uprate power level.

Over the years there have been several dozen uprate applications reviewed and approved by the NRC, as summarized in Table 1-1 [1.1]. Operational problems however have been noted for uprated cores [1.2], which include axial power offset problems, failure to fully insert control rods in high-power/high-burnup fuel assemblies, and deficiencies in core reload neutronics analysis, all of which in some way are partially associated with elevated powers. Issues related to the adequacy of power uprate reviews came to a head in December of 1995, when the agency received anonymous allegations of faulty analyses concerning the power uprate application for the Maine Yankee reactor. The allegation centered on use of non-conservative and inconsistent DBA-LOCA analysis provided by the licensee in support of it's uprate application. An agency self-assessment followed these allegations, the results of were published in the "Maine Yankee Lessons Learned Report" [1.3], which indicated a need for a more comprehensive and coherent review of uprate applications.

This effort is a follow-on to the Maine Yankee self-assessment, but focuses on concerns that potential synergistic processes are not adequately being considered in power uprate applications and the agency's review of such applications. Specifically this report addresses concerns that high core power densities, when combined with potential system/component degradation via plant aging and fuel life extensions to high burnup, may adversely impact safety. This investigation thus centers on an investigation of potential *synergistic* effects, where Webster's definition of synergistic is noted----*the cooperative action of discrete agencies such that the total effect is greater than the sum of the effects taken independently*.

The report is structured as follows. Section 2 describes the scope and format of safety analysis provided in a typical uprate application and agency review procedures for such applications. Section 3 summarizes operational events noted for uprated reactors, which provides an introduction into Sections 4, where potential synergistic concerns are discussed in light of operational events. Section 5 presents report observations and recommendations.

TABLE 1-1: SUMMARY OF POWER UPRATE APPLICATIONS

Plant	Reactor Type	Year Startup	Original Power (MWt)	Yr. Power Uprate	Uprated Power (MWt)	% Power Increase
Oyster Creek	BWR	1969	1690	1971	1930	14.20
Calvert Cliffs-1	PWR	1977	2560	1977	2700	5.46
Main Yankee	PWR	1972	2440	1978	2630	7.79
Millstone-2	PWR	1975	2560	1979	2700	5.47
Fort Calhoun	PWR	1973	1420	1980	1500	5.63
St. Lucie-1	PWR	1976	2560	1981	2700	5.46
Cook-2	PWR	1978	3391	1983	3411	0.59
Duane Arnold	BWR	1975	1593	1985	1658	4.08
St. Lucie-2	PWR	1983	2560	1985	2700	5.47
Salem-1	PWR	1977	3338	1986	3411	2.19
North Anna	PWR	1978	2775	1986	2893	4.25
Callaway	PWR	1985	3411	1988	3565	4.51
Main Yankee*	PWR	1972	2440	1989	2700	10.65
Indian Point-2	PWR	1974	2758	1990	3071	11.35
Fermi-2	BWR	1987	3293	1992	3458	5.01
Wolf Creek	PWR	1985	3411	1993	3565	4.51
Vogel 1&2	PWR	1987	3411	1993	3565	4.51
Peach Bottom-2	BWR	1974	3293	1994	3458	5.01
Susquehanna 1&2	BWR	1983	3293	1994	3441	4.49
Surry 1&2	PWR	1972	2441	1995	2546	4.30
Nine Mile-2	BWR	1988	3323	1995	3467	4.33
Hatch 1&2	BWR	1975	2436	1995	2558	5.00
Limerick-2	BWR	1988	3293	1995	3458	5.01
Limerick-1	BWR	1985	3293	1996	3458	5.01

* Denotes second power uprate, percent power uprate based on original power level.

REFERENCES

1.1 J. S. Miller, *Power Uprate Review*, Scientech, Inc., SCIE-NRC-249-96, (Oct. 1996).

1.2 Institute for Nuclear Power Operations, *Design and Operation Considerations for Reactor Cores*, Institute of Nuclear Power Operations, SOER-96-2, (Nov. 19, 1996).

1.3 Nuclear Regulatory Commission, *Report of the Maine Yankee Lessons Learned Task Group*, Internal NRC Report, (Dec. 1996).

2. POWER UPRATE APPLICATION PROCESS

The power uprate process generally begins with a utility feasibility study to estimate the gain in electrical output and associated increased revenues, versus the cost of the uprate, which would includes any equipment modifications and the costs related to engineering analysis to support the uprate application. Utilities seeking a power uprate must request an amendment to their current operating license from the NRC, which takes the form of a License Amendment request. This documentation should demonstrate that there would be no significant increases in the amount of radiation emitted from the facility at the elevated power level, that any reduction in safety margins for both operational transients and Design Basis Accidents (DBA) would be minimal, and that there are no new or different accidents from those considered in the plant's license basis and FSAR. From a utility perspective, the bulk of the uprate effort generally involves plant-specific engineering and safety evaluations, which detail all affected aspects of the plant at the uprated power level.

Utilities often rely on a generic approach, where the reactor vendor outlines the methods and scope of analysis needed to support a power uprate application, however the uprate request is submitted by the utility. This uprate License Amendment submittal must identify any deviations from the generic approach and provide justification of any plant-specific approaches used in the uprate application. Generally it would include a "no significant hazards" assessment and would address and disposition key licensing issues, such as Regulatory Guides, General Design Criteria, and revisions to Technical Specifications. The uprate application is reviewed by the NRC staff and its findings are reported in the uprate Safety Evaluation Report (SER). Upon agency approval, the applicant utility would then proceed with operational and hardware changes to achieve the increased power, which would normally occur during a refueling outage.

Both General Electric (GE) and Westinghouse (W) have documented generic power uprate methodologies that have been reviewed and approved for use by the NRC. These methodologies are briefly summarized here, indicating the scope of a typical uprate application. This is followed by a discussion of current NRC uprate review practice.

2.1 General Electric-BWR Generic Power Uprate Guidance

Of the four prominent US rector vendors, namely the General Electric Co. (GE) for BWRs, and the Westinghouse Electric Corp. (W), the Babcock & Wilcox Co. (B&W), and the ASEA/Brown-Boveri-Combustion Engineering Co. (ABB-CE) for PWRs, General Electric has provided the greatest power uprate support to utilities using BWRs. The analysis methods and scope of the GE generic power uprate methodology are summarized here, followed by a summary for the more limited support provided by Westinghouse for PWRs. No such generic methodology has been provided by B&W or ABB-CE.

The essential features of the GE-BWR generic power uprate guidance is described in a proprietary GE Topical Report entitled *"Generic Guidelines for GE Boiling Water Reactor Power Uprate"*[2.1], which outlines the scope of analysis and suggested analytical methods to be used for GE-BWR power uprate applications. It documents the FSAR criteria and assumptions that need to be re-evaluated at uprated power levels and specifies calculation methods to address such re-analysis. Table 2.1 provides a description of items covered in generic uprate guidance for GE-BWRs, while additional information is found in Refs. [2.2] and [2.3].

Table 2-1. GE-BWR Power Uprate Guidance

Sample Contents of Plant Specific Power Uprate Report
Specific Licensing Approach for Power Uprate
Specific Assumptions and Bases for Power Uprate Operating Conditions
Specific Assumptions and Bases for ECCS-LOCA Evaluation
Specific Assumptions and Bases for Evaluation
Methods and Assumptions for Control, Instrumentation, & Setpoint Evaluations
Methods and Assumptions for Containment Evaluation
Methods and Assumptions for Radiological Evaluation
Methods and Assumptions for Vessel Components Evaluation
Methods and Assumptions for System Equipment Evaluation
Methods and Assumptions for Piping Evaluation
Specific Assumptions and Bases for Evaluations of Other Aspects of Uprates

In the vernacular associated with GE-BWR uprate applications, the terminology of *"stretch power"* and *"extended power"* are often used. A "stretch power" application involves an increases in power of 5-% or less. It is referred to as a *stretch* increase because the nuclear steam supply system was originally designed for this higher power level. BWR licensee requests for increases in thermal power of up to 120-% of the original FSAR basis are commonly referred to as *"extended power"* applications. This nomenclature stems from the topical report entitled "Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate" [2.2], which provides guidelines for GE-BWR licensees in preparing requests for power increases of this magnitude. Recent BWR uprate applications employing the GE-generic guidelines for *extended* power levels, include the Northern States Power Co. License Amendment Request of July 1996 for the Monticello plant involving a 6.3-% power increase, and the Southern Nuclear Operating Company's uprate request of August 1997 for the Edwin Hatch plant involving an 8-% power increase. Table 2-2 [2.4] provides an overview of submittal information provided by the licensee for Monticello application.

TABLE 2-2. CONTENTS of MONTICELLO-BWR POWER UP-RATE APPLICATION REPORT

TABLE 2-2. CONTENTS of MONTICELLO-BWR POWER UP-RATE APPLICATION REPORT (Continued)

4. **ENGINEERED SAFETY FEATURES**
    - 4.1 Containment System Performance
        - Containment Pressure and Temperature Response
        - Long-Term Suppression Pool Temperature Response
        - Containment Gas Temperature Response
        - Short-Term Containment Pressure Response
        - Containment Dynamic Loads
        - LOCA Containment Dynamic Loads
        - Safety/Relief Valve (SRV) Containment Dynamic Loads
        - Sub-compartment Pressurization
        - Containment Isolation
    - 4.2 Emergency Core Cooling Systems(EM)
        - High Pressure Coolant Injection (HPCI)
        - RHR System (Low Pressure Coolant Injection)
        - Core Spray (CS) System
        - Automatic Depressurization System (ADS)
    - 4.3 ECCS Performance Evaluations
    - 4.4 Standby Gas Treatment System (SGTS)
    - 4.5 Other Systems
        - Post-LOCA Combustible Gas Control
        - Emergency Cooling Water System
        - Emergency Core Cooling Auxiliary Systems
        - Main Control Room Atmosphere Control System
        - Standby Power System
    - 4.6 References

5. INSTRUMENTATION AND CONTROL
    - 5. 1 Nuclear Steam Supply System (NSSS)
        - Control Systems Evaluation
        - Neutron Monitoring System
        - Instrument Set Points
        - RPV High-Pressure Scram
        - High-Pressure RPT
        - Pressure Regulator
        - Safety/Relief Valve
        - Neutron Monitoring System
        - Main Steam High Flow Isolation
        - Steam-line High Radiation Isolation
        - Condenser Low Vacuum Scram
        - Turbine Stop Valve Closure and Control Valve Closure/Scram Bypass
        - Rod Block Monitor and Rod Worth Minimizer
    - 5.2 Balance-Of-Plant (BOP) Power Conversion and Auxiliary Systems
        - Control Systems Evaluation
        - Pressure Control System
        - Turbine Control System
    - 5.3 References

6. ELECTRICAL POWER and AUXILIARY SYSTEMS
    - 6.1 AC Power
        - Generation and Off-Site Power System
        - On-Site Power Distribution System
    - 6.2 DC Power
    - 6.3 Fuel Pool Cooling

TABLE 2-2. CONTENTS of MONTICELLO-BWR POWER UP-RATE APPLICATION REPORT (Continued)

TABLE 2-2. CONTENTS of MONTICELLO-BWR POWER UP-RATE APPLICATION REPORT (Continued)

- 10.2 Environmental Qualifications
  - Quality of Electrical Equipment
  - Inside Containment
  - Outside Containment
- 10.3 Equipment Qualification
  - Mechanical Component Qualification
- 10.4 Required Testing
- 10.5 Individual Plant Evaluations (IPE)
  - Background
  - Power Re-rate Impact on PRA/IPE
  - Evaluations
  - Internal Events PRA -Level 1
  - Internal Events PRA -Level 2 (Containment Analysis)
  - External Events
  - Summary of Results
  - Comparison to PSA Application Guide
  - Comparison to Original IPE Submittal to NRC
  - Conclusions
- 10.6 References

11. LICENSING EVALUATIONS
- 11. 1 Evaluation of Other Applicable Licensing Requirements
  - NRC and Industry Communications
  - Plant-Unique Items
  - Commitments
  - Safety Evaluations
  - Temporary Modifications
  - Updated Safety Analysis Report Changes
  - NRC Safety Evaluation Reports
  - Emergency Operating Procedures
- 11.2 Impact on Technical Specifications
- 11.3 Environmental Assessment
- 11.4 Significant Hazards Consideration Assessment
- 11.5 References

2.2 Westinghouse-PWR Generic Power Uprate Guidance

Westinghouse, Combustion Engineering, and Babcock & Wilcox represent the three major PWR vendors, however only Westinghouse has provided guidance for power uprate applications of its reactors. This guidance is documented in a Westinghouses Topical Report, WCAP-10263, *"A Review Plan for Uprating the Licensed Power of a Pressurized Water Reactor Plant"* [2.5]. WCAP-10263 is much less ambitious and detailed than the GE generic guidance for BWRs. It was written in the early 1980s, as a starting point for discussions with NRC to define the basic ground rules and criteria that would be acceptable for evaluating PWR uprate applications. Ever since that time WCAP-10263 has served as a basic blueprint for PWR uprate applications, not only for Westinghouse reactors but for B&W and CE reactors as well. It enumerates the types of safety evaluations and component design reviews that should be included in any PWR uprate application.

WCAP-10263 lists the types of accidents and transients that need to be analyzed for PWRs, which mirror closely analysis covered in the FSAR that is used in an initial license application. Suggested analysis include design limiting events to establish for DNB margins at the new power level, and a re-analysis of reactivity excursions, ECCS capability, peak reactor pressure and core heatup analysis. WCAP-10263 also provides guidance for demonstrating the adequacy of plant systems and components at the uprated power. Typical plant transients to be considered in an uprate application are listed in Tables 2.3 and 2.4. Calculational models and criteria are not specified in WCAP-10263, thus the PWR generic guidance is much less ambition in scope than that supplied for BWRs by GE.

Recent PWR power uprate applications, employing guidance from the Westinghouse Uprate Methodology, include the Virginia Power Co. License Amendment Request of August 1994 for the Surry-1& 2 units, and the Wolf Creek Nuclear Operating Corp. uprate request of December 1992 for the Wolf Creek plant. Table 2-5 [2.6] provides an overview of the scope of the Surry submittal.

TABLE 2-3. Summary of W-PWR 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/Unloading at 5-% full power/min
3. Step Load Increase/Decrease at 10-% 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 for (a) Primary Side, (b) Secondary Side, and (c) Primary Side Leak Test

TABLE 2-4. Typical Accident Analysis

- Uncontrolled RCC Assembly Withdrawal
- RCC Assembly Misalignment
- Chemical & Control System Malfunction
- Loss of Reactor Coolant Flow
- Startup of an Inactive Coolant Pump
- 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
- Rupture of a Control Rod Drive Mechanism Housing
- Reactor Coolant System Pipe Rupture

TABLE: CONTENTS OF SURRY-PWR POWER UP-RATE APPLICATION REPORT

TABLE: CONTENTS OF SURRY-PWR UP-RATE APPLICATION REPORT (Continued)

2.3 NRC Power Uprate Review Process

Utilities seeking power uprate approvals from NRC will submit a License Amendment Report,which documents the utilities safety case for the uprate. This licensing report is reviewed by the NRC staff, with findings reported in a *Safety Evaluation Report (SER)*. The NRC review generally centers on an evaluation of the licensees analysis of the impact of the power uprate plant operations and safety, including the impact of any changes in plant operation or equipment stemming from the uprate. For application approval, the staff would need to conclude that there would be no significant increases in the amount of effluents or radiation emitted from the plant, a nil reduction in safety margins for both operational transients and Design Basis Accidents (DBA), and that no new or different accidents would occur at the increased power than those considered in the original Safety Analysis Report (FSAR) and license basis.

It should be noted that the agency does not have a *Uprate Standard Review Plan,* which would standardize review procedures, as well as guidance and acceptance criteria, in a manner similar to *Standard Review Plan* (SRP, NUREG-0800) for review of the original plant FSAR. Lack of standardized uprate review procedures has lead to differences in the scope of reviews for various uprate applications. In some cases, the staff review has centered on assurance that the plant specific analysis are consistent with generic approaches. In other cases, a particular plant feature may be sufficiently similar to that of a plant that has received prior uprate review, so that a "de novo" review of that particular aspect or feature may not be conducted by the staff.

Table 2-8 summarizes the topic areas documented in the SER [2.7] for the Wolf Creek-PWR uprate application [2.8], which is typical of the scope of most uprate SERs. Inspection of this and other SERs, reveals essentially nil plant-specific analysis or independent code calculations performed by the NRC staff in its review. Rather the SER provides only general statements to the effect that the staff reviewed the licensees assessment and found it to be satisfactory or unsatisfactory. In general, no supporting/independent analysis are provided by the staff in the SER, nor are acceptance criteria usually given upon which staff conclusions were reached. The following are excerpts of staff review findings found in a typical uprate-SERs, in this case for the Wolf Creek review:

Emergency Core Cooling System (ECCS): "From the licensee's study, no adverse impact to ECCS operability or vulnerability to single failure due to the re-rated conditions was identified. The licensee submitted revised ECCS performance analyses in support of Amendment 61, which justified various changes associated with Cycle 7 operation. The licensee performed large and small break analyses at the limiting re-rate conditions and determined that all acceptance criteria continued to be satisfied. The NRC staff has reviewed the licensee's analyses and concludes that the ECCS analyses referenced in support of the re-rate conditions continues to be in compliance with 10CFR50.46 and App. K. The Wolf Creek ECCS is, therefore, acceptable for operation at the re-rated conditions."

Main Steam System: "The main steam system dissipates energy generated by the reactor core to the turbine generator and auxiliary steam loads, the main condenser via the steam dump valves, or to the atmosphere via atmospheric relief valves or main steam safety valves. Isolation of the main steam system is achieved by the main steam isolation valves and main steam bypass isolation valves. The licensee evaluated the capability of the main steam system components to perform their design functions under the proposed re-rate conditions. The licensee determined that the existing set-points and capacity of the main steam safety valves are adequate to prevent exceeding 110-% of design pressure of the main steam system under the most limiting transient. The set-point and capacity of the atmospheric relief valves were found to remain adequate to control the design load shed of 10-% rated thermal power. In addition, the atmospheric relief valves were found to have adequate capacity to achieve a 50 F/hr cooldown if the main condenser was unavailable. The main steam isolation valves were evaluated to ensure the valves will continue to perform their isolation function under the maximum differential pressure conditions and within the time limits assumed in the safety analysis. The staff concludes that the existing main steam system components are adequate to perform their safety functions under the re-rated plant conditions."

Main Feedwater: "The main feedwater system delivers feedwater, at the required pressure and temperature, to the four steam generators. The safety-related portions of the system ensure isolation capability and provide a path to permit the addition of auxiliary feedwater for reactor cooldown following design basis transients. The licensee's evaluation shows that the existing design basis for the main feedwater isolation valves and main feedwater bypass isolation valves is not significantly affected by operation at the re-rate conditions. The piping configurations associated with the feedwater and auxiliary feedwater systems do not change as a result of the re-rate conditions. The ability of the auxiliary feedwater system to perform its heat removal function was addressed by the licensee. The staff finds that the safety functions of the feedwater system will continue to be satisfied during operation at the re-rate conditions."

In each of the above examples, no agency analysis or independent code calculations are cited to support staff conclusions. As also noted in the Maine Yankee Lessons Learned report [2.8], uprate-SERs do not generally specify the NRC staff member which performed a specific part of the review, the scope of the subject matter reviewed by that staff, how the review was accomplished, and acceptance criteria for the conclusions reached. Such information is however required in the review of the original plant FSAR, as specified Standard Review Plan (SRP, NUREG-0800) for. Of particular note are standardization of acceptance criteria. In the FSAR-SRP the technical bases for the acceptance criteria are specified, and the SRP typically specifies the solutions and approaches determined to be acceptable by the staff in dealing with a specific safety problem or safety-related equipment design. These solutions and approaches are codified in a form so that staff reviewers can rely on uniform and well-understood positions for review of all plants. Some standardization of requirements, evaluation tools, and acceptance criteria would likewise appear desirable for review of power uprate applications.

Table 2-8. Content of Wolf Creek Uprate Safety Evaluation Report (SER)

Section	Description of Content
Introduction	1 pg: Brief overview of uprate power level and affected plant conditions
Nuclear Steam Supply	3 p.s.: Review of Licensee assessment of applicable NSS system codes/standards, core thermal conditions, over-pressure protection assurance, AUX and residual heat removal capacity, and licensee ECCS analysis. Check of consistency of source term analysis with original licensing basis.
Safety Cooling Systems	1.5 p.s.: Review of Licensee assessment of ultimate heat sink capability at uprated power, as well as that for essential service water system, spent fuel pool, and component cooling water system.
Balance of Plant	2 p.s.: Review of licensee assessment of adequacy of turbine over-speed protection, valve set-points and flow capacities, and feedwater system to provide its heat removal function
Containment Analysis	2 p.s.: Review of licensee containment integrity analysis for uprated power and under LOCA-DBA conditions
Plant Structural Analysis	4 p.s.: Review of licensee evaluations of the structural integrity of reactor vessel, piping, control rod mechanisms, steam generator, reactor coolant pumps and pressurizer at the uprated power level remain bounded by original design basis analysis or that any changes are acceptable.
Miscellaneous	3 p.s.: Review of licensee evaluations pertaining to radiological dose to equipment, adequacy of main generator/transformers, Rad waste and HVAC systems, and threat of internal flooding at uprated power level.
Tech Specs	1 pg: Review of licensee proposed changes in plant operating procedures and technical specifications

2.4 Maine Yankee Lessons Learned

In December of 1995 the agency received an anonymous allegation of faulty analyses concerning two power uprates granted for the Maine Yankee reactor (see Table 1-1). The allegations asserted that the licensee knowingly performed faulty analyses of the emergency core cooling system (ECCS) and the containment in support of two power uprate approvals granted by the agency. The fact that the conditions asserted in these allegations were not identified by the NRC staff during its review, implied a weaknesses in the agency's power up-rate review and approval processes. The agency responded to these allegations by initiating a self-assessment effort, which was documented in the *"Report of the Maine Yankee Lessons Learned Task Group"* [2.9].

The subsequent investigation into the Maine Yankee allegations indeed showed that the plant specific application of RELAP-5 code for the small break LOCA portion of the ECCS uprate analyses, did not conform to the requirements of 10CFR50.46 and 10CFR50-Appendix K. The NRC allegations review team performed an assessment of the code review process, the code modification process, and staff follow-up of vendor and licensee code implementation. The review team found a need for improved consistency and uniformity in the uprate review process and the need for more formal staff guidance in the review of code applications. The allegations review team also noted concerns regarding the licensee's estimates for containment pressures at the uprated power level, and likewise identified mechanical components for which operability at the upgraded power level could not be confirmed. A more general conclusion stemming from the Maine Yankee self-assessment effort was the identification of the need for standardization of license amendment reviews for power uprates and integration of technical review conclusions into licensing basis documents.

In addition to the internal self-assessment effort, an independent assessment of the adequacy and consistency of agency uprate review procedures was conducted by Scientech Inc. [2.10], under contract to NRC. In this study, 31 power uprate applications were reviewed. It was found that 9 of the uprate applications were not reviewed by the staff, because the FSARs for these plants had been written assuming the power levels to which the plants were later uprated. For these plants the Commission granted a provisional operating license at a specific thermal power level less than the design basis rating, until its performance could be evaluated. Later, the Commission granted a full-term operating license at the design basis power rating. For these plants the NRC staff simply certified that analyses in the FSAR were performed at the full uprated power level. The remaining 22 power uprate applications were however reviewed by the staff, with staff uprate review findings reported in a plant specific *Safety Evaluation Report (SER).* Scientech reviewed each of these 22 submittal applications and associated SER, as well as licensee responses to staff requests for additional information.

A primary conclusion of the Scientech Inc. study [2.10] was that the scope and depth of the power uprate reviews performed by the NRC staff varied substantially. Although all uprate applications were reviewed with respect to design basis accident and transient events, nuclear design and core thermal-hydraulics Design, containment performance, and balance of plant, the Scientech study indicated that the review of a number of uprate applications did not go beyond the review of these limited topic areas. It was also noted that the scope of staff review efforts tended to increase with time, that is reviews performed in the 1990s were more extensive than earlier reviews, which largely reflects the fact that later uprates had the benefit of the GE and Westinghouse guidance for uprate submittals. With regards to questions of uprate application review consistency, the Scientech study found that often the NRC Safety Evaluation Report (SER) did not contain sufficient information as to whether the analysis at the uprated power was done with the same code used in the original FSAR analysis or with another code. For applications where the licensee performed first-time analyses using a new code, there was no evidence that an evaluation was performed to determine the effect on safety margin using the new code. The Scientech team likewise noted that although a Safety Evaluation Report (SERs) was issued for each of the 22 reviewed uprate applications, many SERs only identified the NRC project manager for the review, while most did not list the various contributors to these reviews.

As a result of the Maine Yankee self-assessment effort, a number of agency actions were undertaken, which included NRR efforts to determine if any previously approved uprate applications should be re-evaluated and to what extent. An internal report was prepared by the agency's PSA (Probabilistic Safety Assessment) branch [2.11], which concluded that a plant-by-plant IPE/PSA (Individual Plant Examination/Probabilistic Safety Assessment) screening of uprated plants was not practical because of the minor impact on CDF (Core Damage Frequency) and LERF (Large Early Release Fraction) for power uprates in the 5-8 percent range. Thus no re-opening or re-evaluation of any prior uprate applications was conducted.

The Maine Yankee Lessons Learned report [2.9] also suggested that updates to FSAR be made following approval of a power uprate, to reflect changes in operational conditions, equipment modifications, and technical specifications accurately reflect plant changes resulting from uprated conditions.

One of the more important conclusions of the Maine Yankee review was the need to develop a Standard Review Plan for Power Uprates. Although the agency has a Standard Review Plan (SRP) for original applications, the SRP essentially guides the technical review of specific systems and parts of an application for a construction permit or operating license. The agency does not have a formalized procedure for handling an application for a power uprate or other applications for a license amendment. No guidance is in place, such as an NRR office letter, which specifies the scope and detail of an uprate review, much less acceptance criteria to be used for making judgements on the adequacy of the uprate analysis. The Maine Yankee Lessons Learned report [2.9] suggested that any power uprate review plan or procedure include the following elements:

l) specify the analytic codes to be used in the support of an application for a power uprate and specify how their use should be reviewed,

2) include guidance on review of licensee analyses and technical specifications,

3) alert the reviewer to the need to consider all cumulative potential decreases in safety margin that have occurred over the years from successive plant and procedure modifications, including that stemming from the proposed power uprate,

4) identify the technical review branches that should contribute to the review,

5) specify what information should be provided by the Project Manager on the history and open items for the plant,

6) specify what follow-up is needed regarding license conditions and FSAR update, and

7) clearly indicate the differences between review requirements for plants using GE-BWR generic guidance for "stretch" power uprate applications (105-% original FSAR power level), versus applications for plants using GE-BWR generic guidance for "extended" power uprate applications (up to 120-% original FSAR power level).

At this time the agency has not yet undertaken serious efforts to develop a Standard Review Plan for review of Power Uprate submittals, although it was stated in Refs.[2.12 and 2.13] that as higher priority items are brought to closure, staff resources would be assigned to develop such a standard review plan.

Although the Maine Yankee allegations prompted some NRC urgency regarding the adequacy of uprate reviews, it is noteworthy that a number or recent operational incidents have occurred involving plants that received power uprate approvals; which are discussed in the following section.

2.5 REFERENCES

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2.4. General Electric Company, *Power Rerate Safety Analysis Report for Monticello Nuclear Generating Plant,* General Electric Co. Licensing Topical Report, NEDC-32546P, (July 1996).

2.5 R. H. McFetridge, R. T. Marchase, and R. H. Faas, *A Review Plan for Uprating the Licensed Power of a Pressurized Water Reactor Power Plant*, Westinghouse Electric Corp., WCAP-10263, (Jan. 1983).

2.6 Virginia Power Corp, *Surry Power Station Core Uprate Licensing Report*, NRC Docket Number 50-280, (Aug. 1994).

2.7 U. S. Nuclear Regulatory Commission, *Safety Evaluation Report (SER) for Amendment of Facility Operation License for the Wolf Creek Generating Station*, NRC Docket Number 50-482, (Nov. 10, 1993).

2.8. Wolf Creek Nuclear Operating Corp., *Power Uprate Report for the Wolf Creek Generating Station,* NRC Docket Number 50-482, (Dec. 1992).

2.9. P. Cota, A. Cubbage, D. Dorman, and R. Caruso, *Report of the Main Yankee Lessons Learned Task Group*, Internal NRC Report, (December 1996).

2.10. J. S. Miller, *Power Uprate Review*, Scientech, Inc., SCIE-NRC-249-96, (Oct. 1996).

2.11. R. J. Barrett, *SPSB Contribution to the Maine Yankee Lessons Learned Action Plan*, NRC Memorandum to J. A. Zwolinski, (Oct. 7, 1998).

2.12. G. M. Holahan, *Maine Yankee Lessons Learned Action Plan*, NRC Memorandum to J. A. Zwolinski, (May 8, 1998).

2.13. S. J. Collins, *Status of NRR Staff Actions Resulting from the Independent Safety Assessment of Maine Yankee Atomic Power Company*, NRC Memorandum to L. J Callan, (Jan. 27, 1998).

3. EVENTS NOTED FOR POWER UPRATES

This section presents a brief overview of operational events and other incidents noted for plants that have received uprate approvals by the agency. For each case a brief description of the event or problem is first given, followed by the licensee uprate submittal information and NRC assessment of that information that most pertains to the event or problem. Lastly a assessment of adequacy of uprate review process to deal with the event or problem is noted. The assessment of the adequacy of the uprate review process is based solely on a review of information documented in the licensee's Power Uprate Licensing Report and the associated NRC Safety Evaluation Report (SER). It is also cautioned that in some cases multiple effects may have contributed to the event or problem, one of which being power level.

3.1 Maine Yankee LOCA Analysis

Event Description: As discussed in the previous section, in December of 1995 the agency received anonymous allegations of faulty design basis LOCA analyses for two power uprate approvals granted for the Maine Yankee reactor. The allegations appeared to be made by a knowledgeable party and asserted that the licensee, the Maine Yankee Company, performed small-break LOCA analysis using NRC accepted critical flow (Moody correlation) and decay heat models, where calculated fuel rod cladding temperatures exceeded the regulatory limit 2200 F due to insufficient Emergency Core Cooling System (ECCS) capabilities for the plant at the uprated power levels. The allegations asserted that the results of these calculations were not submitted to the NRC; rather the Maine Yankee submittal contained code predictions that removed conservatism in both the decay heat and critical flow models.. The allegations also noted that licensee's analysis for large-break LOCAs indicated predicted containment pressures that exceeded the containment design pressure. Maine Yankee did not reveal this information in its submittal, and likewise did not upgrade the ECCS in light of its in-house information. Such faulty analysis were not identify by NRC staff during its reviews for both the 1989 uprate from 2630MWt to 2700 MWt, and the pre-1989 uprate from 2400MWt to 2630MWt

An NRC allegations review team was formed in early 1996 to investigate these allegations. The review team found that the licensee's submittal information of plant specific application of RELAP-5 code for the small break LOCA portion of the ECCS uprate analyses indeed did not conform to the requirements of 10CFR50.46 and 10CFR50-Appendix K. The NRC allegations review team performed an assessment of the code review process, the code modification process, and staff follow-up of vendor and licensee code implementation. The allegations review team also noted concerns regarding the licensee's estimates for containment pressures at the uprated power level, and likewise identified mechanical components for which operation at the upgraded power level could not be confirmed. A more general conclusion stemming from the Maine Yankee self-assessment effort was the identification of the need for standardization within NRC of license amendment reviews for power uprates and integration of agency technical review conclusions into licensing basis documents.

Licensee Application: The following statements are abstracted from utility's uprate Licensing Report [Ref. 3-3] related to small-break LOCA analysis (see Section 5.5.5.3)

The small-break LOCA analysis performed by Combustion Engineering for Cycle 4 considered a spectrum of cold leg breaks varying in size from 0.1 to 0.5 ft^2. Results showed that the limiting break size is the 0.5 ft^2 break, with a peak clad temperature of 1348 F, well below the acceptance criteria of I0CFR50.46. A demonstration analysis of the limiting break performed for Cycle 5 (1) utilizing YAEC methodology yielded a peak clad temperature of 1230 F, well below the I0CFR50.4 acceptance criteria and the Maine Yankee large-break results. In that analysis, a 68% peak top skew design shape and a linear power level of 16 kW/ft were used. The analysis predicted a short period of core uncovery and resultant cladding heatup. Thus, small break LOCAs for Maine Yankee were shown to be non-limiting.

The results of previous analyses are applicable to Cycle 11 operation at an uprated power level of 2700 MWt because they are determined primarily by the decay heat; values which are insignificantly impacted by the increased power level. Additionally, slight differences in Cycle 11 and Cycle 5 system configuration would not significantly affect the PCT (Peak Cladding Temperature), which was predicted to be well below the 10CFR50.46 criteria. Hence, the minor system changes will not make the small-break LOCA a limiting scenario.

NRC Safety Evaluation Report: The following statements are abstracted from the Maine Yankee SER [Ref. 3-4], indicating the level of staff effort with regards to review of the licensee's small-break LOCAs analysis (see Section 5.9 of the SER):

> For Cycle-11 operation at 2630 MWt, the break spectrum analysis performed for Cycle 10 was found to be applicable as approved by the staff . However, for the uprate to 2700 MWt, the Cycle 10 analysis was no longer applicable. Therefore, a new break spectrum calculation for operation in Cycle-11 at 2700 MWt was performed. This new calculation used the same methodology and assumptions as that in the Cycle-10 analysis, except the core power level was increased to 2700 MWt, the Cycle-11 reactor kinetics parameters were used, and the staff-approved steam-cooling model was used.
>
> For each of the limiting breaks, a LOCA calculation was performed with input date specifically for Cycle 11 operation at 2700 MWt. The results of the analysis for each axial power shape indicate that the cladding temperature, cladding oxidation, and hydrogen generation values are in compliance with 10CFR50.46/Appendix-K criteria.
>
> Previous analyses have shown that small break LOCA for Maine Yankee are non-limiting. The results of these previous analyses are determined primarily by the decay heat values, which are insignificantly impacted by the increased power level. In addition, since the peak clad temperature and other parameters were calculated to be well below the 10CFR50.46 criteria, it would not be significantly affected by either slight differences in core configurations between cycles or power level. The staff, therefore, concludes that the results of previous small break LOCA analyses for Maine Yankee are applicable to Cycle 11 operation at 2700 MWt.

Discussion: Although agency investigations into the Maine Yankee allegations corroborated the contention of a deliberate submittal of faulty LOCA analysis provided by the licensee, as well as faulty applicant estimates for containment pressures at the uprated power level, such inadequacies were not revealed in the initial agency uprate review [Ref. 3-4], indicating a less than satisfactory review and approval processes. The NRC Maine Yankee review team recommendation for standardization of license amendment reviews for power uprates and the need for consistency and uniformity of code applications in uprate submitttals is well founded.

As discussed below, a review of other events for uprated plants indicates to this reviewer the need for agency capabilities to perform independent/in-house thermal-hydraulic and neutronic code analysis, to verify the accuracy of licensee uprate submittal calculations. At this point in time, essentially nil quantitative analysis by the NRC review staff is done as part of the uprate review and approval process. Such agency in-house capabilities and practice may have brought to light faulty Maine Yankee licensee submittal information during the uprate review process, rather then being brought to light through third party allegations.

3.2 Wolf Creek/North Anna Control Rod Events

Both the Wolf Creek and North Anna PWR plants received power uprate approvals in the range of 4-5% and experienced subsequent control rod insertion problems [Refs. 3-5,3-6] in high burnup fuel assemblies (i.e. > 45 GWD/t-U). These events are thus examined in the light of potential synergistic control rod/fuel burnup/power uprate effects.

Plant	Original Power	Yr. Power Uprate	Uprated Power	% Power Increase
Wolf Creek	3411 MWt	1993	3565 MWt	4.5-%
North Anna-1	2775 MWt	1986	2893 MWt	4.3-%

3.2.1 Wolf Creek Event

Event Description: Five control rods at Wolf Creek PWR plant failed to properly insert during a plant trip on January 30, 1996. All of the affected control rods involved Westinghouse VANTAGE-5H fuel assemblies with burnups greater than 47,6000 MWD/t-U. As indicated, the Wolf Creek plant received agency approval in 1993 for a 4.5-% power uprate from 3411MWt to 3565MWt. Root cause analysis [Ref. 3-7] revealed that the control rod insertion problems were caused by fuel assembly guide thimble tube distortion resulting from excessive compressive loading. The compressive loading was caused by excessive irradiation induced growth of the Zircaloy thimble tubes at high power/high-burnup core locations.

Licensee Application: The following statements are abstracted from utility's Uprate Licensing Report [Ref. 3-8] related to control rod performance at uprate power conditions:

The effects of uprating the allowable thermal power and increasing the core peaking factor limits on the nuclear design bases and methodologies for Wolf Creek Generating Station have been evaluated.

The uprated allowable thermal power is 4.5% more than the currently licensed power level. The effects of the allowed thermal power and associated fuel and moderator temperature changes on core physics characteristics are small and are explicitly modeled in the neutronics models. The specific values of core safety parameters, e.g., power distributions, peaking factors, rod worth, reactivity coefficients, are primarily loading pattern dependent. The variations in the loading pattern dependent safety parameters are expected to be typical of the normal cycle to cycle variations for the standard fuel reloads.

In summary, the increase in allowed thermal power from the current level will not reduce the margin of safety in the current Wolf Creek Safety Analysis Report (SAR) nuclear design bases. However, the design bases will be modified due to the increases to the peaking factor limits and allowed thermal power.

No changes to the nuclear design philosophy or methods are necessary because of the increased allowable thermal power or the use of increased peaking factors. The reload design philosophy includes the evaluation of the reload core key safety parameters which comprise the nuclear design dependent input to the USAR (Updated Safety Analysis Report) safety evaluation for each reload cycle. These key safety parameters will be evaluated for each reload cycle. If one or more of the parameters fall outside the bounds assumed in the safety analysis, the affected transients will be re-evaluated and the results documented in the reload safety evaluation for that cycle.

3.1.2 Core Thermal-Hydraulic Evaluation: This section describes the calculational methods used for the thermal-hydraulic analysis, evaluation of the departure from nucleate boiling (DNB) performance, and the hydraulic compatibility during the transition from mixed fuel cores to an all VANTAGE-5H with intermediate flow mixers (IFM) core. Based on minimal hardware design differences and prototype hydraulic testing of the fuel assemblies, it is concluded that the standard VANTAGE-5H fuel assembly and VANTAGE-5H fuel assembly with IFM grids are hydraulically compatible. Table 3.1.2-1 provides a summary of the thermal-hydraulic design parameters for the Wolf Creek plant CGS that were used in this analysis. The thermal-hydraulic design for the upgraded fuel product was analyzed for an increase in the design limit value for the nuclear enthalpy rise hot channel factor (F_h) from 1.55 to 1.65. This increase is achieved by removing unnecessary conservatism in the design through the use of an improved critical heat flux correlation and improved analysis methodologies as described in the following sections. The thermal-hydraulic design criteria and methods remain the same as those presented in the Wolf Creek SAR with the exceptions noted in the following sections. All of the current USAR thermal-hydraulic design criteria are satisfied.

3.1.2.3 Hydraulic Compatibility: Fuel assembly lift forces are defined as the net upward force acting on the assembly due to interaction with coolant flow, excluding fuel assembly weight and buoyancy. Fuel assembly lift forces are used in the design of fuel assembly hold-down springs and reactor vessel internals.

Lift forces are calculated at hot full power, cold startup and hot pump over-speed. Designing to these conditions ensures that the hold-down spring design criterion is met.

The Wolf Creek Generating Station will transition to VANTAGE-5H fuel assemblies with IFM grids (intermediate flow mixers) from a core consisting of VANTAGE-5H (without IFM grids) and Standard (STD) fuel assemblies. Consequently, lift forces for all three fuel types were evaluated. Based on this evaluation, it was concluded that the hydraulic load on the hold-down spring and the core internals for STD, VANTAGE-5H, and the VANTAGE-5H fuel assemblies with IFM grids is acceptable.

3.1.2.5 Effects of Fuel Rod Bow on DNBR: The phenomenon of fuel rod bowing must be accounted for in the DNBR (DNB ratio) safety analysis of Condition I and Condition II events. In the IFM grid region of a VANTAGE-5H fuel assembly with IFM grids, the grid-to-grid spacing is approximately 10 inches compared to approximately 20 inches in the current fuel assemblies in the Cycle 6 core. Using approved methodology, the predicted channel closure in the 10 inch spans in the VANTAGE-5H assemblies with IFM grids will be less than 50%. Thus, no rod bow penalty is required in this region. In the spans below the IFM region of the VANTAGE-5H assemblies with IFM grids and for the resident fuel, rod bow is accounted for in available DNBR margin as summarized in section 3.1.2.2.2.

The maximum rod bow penalties accounted for in the design safety analyses are based on an assembly average burnup of 24,000 MWD/t-U, as approved by the Commission. At burnups greater than 24,000 MWD/T-U, credit is taken for the effect of F_h (nuclear enthalpy rise hot channel factor) burndown, due to the decrease in fissionable isotopes and buildup of fission product inventory. No additional rod bow penalty is required.

3.1.2.6 Fuel Temperature Analysis: The 0.374 inch O.D. fuel rod used in the VANTAGE-5H fuel assembly with IFM grids is the same as that used in the Standard and VANTAGE-5H fuel assemblies (without IFM grids) resident in the core. Fuel performance evaluations are completed for each fuel region to demonstrate that the design criteria will be satisfied for all fuel regions under the planned operating conditions for each reload core. Fuel rod design evaluations are performed using approved models. There is no change in the fuel temperature design criteria used in the safety analysis calculations between the fuel types resident for cycle 6 and the VANTAGE-5H assemblies with IFM grids to be loaded for cycle 7.

3.1.2.7 Transition Core Effects: The fuel to be loaded for cycle 7 has IFM grids located in spans between mixing vane grids in the upper region of the fuel assembly. The resident fuel, both the Standard and VANTAGE-5H assemblies, does not feature these intermediate grids. The additional grids introduce localized flow redistribution from the VANTAGE-5H assemblies with IFM grids assemblies into the Standard and VANTAGE-5H assemblies at axial zones near the IFM grid positions in a transition core. Between the IFM grids, flow returns to the VANTAGE-5H assemblies with IFM grids due to the tendency for velocity equalization in parallel open channels. This localized flow redistribution actually benefits the Standard and VANTAGE-5H assemblies. This benefit more than offsets the slight mass flow bias due to velocity equalization at non-grid locations. Thus, the analysis for a full core of these fuel assembly types remains appropriate for that fuel in a transition core.

Transition cores are analyzed as if they were a full core of one assembly type, VANTAGE-5H with IFM grids in this application. A transition core penalty is then applied to the thermal-hydraulic design analyses to account the impact of the flow redistribution. For VANTAGE-5H with IFM grids, the transition core penalty is a function of the number of VANTAGE-5H with IFM grid assemblies present in the core and is determined using approved methodologies. The transition core penalty for Cycle 7 operation has been established at 12.0%. This penalty is included in the safety analysis limit DNBR such that sufficient margin over the design limit DNBR exists to accommodate the transition core penalty along with other appropriate DNBR penalties.

3.1.2.8 Conclusions: The thermal-hydraulic evaluation of the fuel upgrade and peaking factor increase for the Wolf Creek Generating Station has shown that 17x17 STD fuel assembly, VANTAGE-5H, and VANTAGE-5H assemblies with IFM grids are hydraulically compatible and that the DNB margin gained through the use of the Statistical Core Design methodology and the WRB-2 critical heat flux correlation is sufficient to allow an increase in the design F_h (nuclear enthaply rise hot channel factor) from 1.55 to 1.65. More than sufficient DNBR margin exists in the safety limit DNBR to cover the rod bow and transition core penalties. All thermal-hydraulic design criteria are satisfied.

NRC Safety Evaluation Report: The associated NRC evaluation for control rod performance is essentially limited to the following statement from the SER [Ref. 3-9 , see pg 10]:

The licensee evaluated the adequacy of the Control Rod Drive Mechanisms (CRDMs) by comparing the design bases input parameters with the operating conditions for the proposed re-rate. The licensee stated that the re-rate conditions would have an insufficient impact on the original design basis analysis for the CRDMs. The staff has reviewed the licensee's evaluation and concurs with the Licensee's conclusion that the current design of the CRDMs would not be impacted by the re-rate (Pg 10, SER)

Discussion: No mention of potential changes in fuel operational conditions, margins, flow redistribution or other fuel behavior effects were discussed in the SER. Potential synergistic effects due to the combined effects of higher power and higher fuel burnup were not discussed in either the Licensee Safety Uprate Report, nor the agency's SER. It is thus concluded that synergistic high power/high-burnup effects were not addressed.

3.2.2 North Anna Event

Event Description: Control rod problems likewise occurred at the North Anna-1 plant on Feb. 21, 1996, where retrieval difficulty was noted for two new control assemblies being temporarily stored in the plant's spent fuel pool [Ref. 3-5]. The two affected control assemblies were being stored in spent Westinghouse VANTAGE-5H fuel assemblies, which had achieved burnups of 47,700 MWD/t-U and 49,600 MWD/t-U. The control assemblies were eventually removed using the rod assembly handling tool in conjunction with the plant's bridge crane hoist. To ascertain the cause of binding, the two affected control rod assemblies were subsequently inserted into other low-burnup fuel assemblies using normal assembly handling techniques, where no additional binding was observed. However, difficulty was experienced when a different/new control rod assembly was inserted into the two high-burnup fuel assemblies that were associated with the binding problem. On this basis the licensee determined that the cause of the binding was related to fuel rods in assemblies with high burnup and not the control rod assemblies themselves. Subsequent control rod drag testing data indicated a correlation of control rod drag force to assembly burnup and a significant increase in drag force at assembly burnups greater than 45 GWD/t-U.

Licensee Application: The following statements are abstracted from utility's Uprate Licensing Report [Ref. 3-10] related to control rod performance at uprate power conditions:

> Review of the control rod drive mechanism design showed that operating conditions for 2910 MWt operation are bounded by the original thermal and structural design analysts.

No additional statements are found with respect to control or fuel rod analysis for the requested power uprate.

NRC Safety Evaluation Report: The associated NRC evaluation for control and fuel rod performance is essentially limited to the following statement from SER [Ref. 3-11, see pg 4) :

> The licensee re-analyzed the consequences of postulated control rod ejection accidents at the beginning and end-of-life. The results were not significantly different from those of the FSAR and remain below the staff's acceptance criteria for maximum fuel sensible heat and percent fuel melting. Boron dilution events were not re-analyzed since the course of these events would not be affected by the power upgrade.

Discussion: No mention of other potential changes in control rod or fuel performance, or synergistic high-burnup/high power density effects were assessed in the licensee application or discussed in the agency's SER.

The failure of control rods to fully insert in several PWR high burnup assemblies is one of the notable problems encountered in core reloads using extended-life fuel assemblies operated at relatively high core power locations. The problem was first noticed at South Texas-1 plant on Dec. 1995, when three control rods failed to fully drop into the core as the unit tripped from 100-% power. The control rods in question involved a Westinghouse XLR type fuel assembly, at a burnup level exceeding 42,880 MWD/t-U. Although the South Texas-1 unit was operating at its original certification power level, it is noted that this unit is a relatively new plant (began commercial operation in 1988) operating at power densities equivalent to older plants that have uprated to higher power levels. Although not fully understood, irradiation induced growth of Zircaloy thimble tubes appears to be the prime culprit for control rod binding problems, for which synergistic power-level/burnup effects can be expected. This is because irradiation associated Zircaloy growth is dependent on both neutron fluence, a power level effect, and total neutron exposure or dose, which is a burnup related effect.

3.3 Surry-2 Main Feedwater Pipe Rupture

Event Description: The main feedwater pipe break at Surry-2 on Dec. 9, 1986 provides an *indirect* indication of potential synergistic effects between plant aging phenomena and increased coolant flow/turbulence that may accompany power uprate conditions. During at-power operation a main feedwater pipe ruptured at the Surry-2 plant, resulting in four site-worker fatalities due to release of scalding steam from the ruptured pipe [Refs. 3-12, 3-13]. Post-accident investigations [Ref. 3-14] revealed feedwater pipe thinning and catastrophic rupture due to combined corrosion/erosion effects. Although the Surry-2 plant was operating at its original power rating at the time of the event, a linkage to uprated power conditions is made here since often power uprates involve an increase in feedwater flow conditions, where erosion effects would be expected to be exacerbated at higher flow rates.

When the two Surry units first entered operation in the early 1970s, untreated water from the James River was used as secondary-side coolant. Water impurities and high oxygen content corroded the outer surface of the steam generator tubes, forcing the utility to replace the steam generators for both units in the early 1980s. To protect the new steam generators, a water treatment system was added to the secondary-side demineralization system to affect control of oxygen to low levels. Although the treated water had the desired effect of reducing tube wall corrosion in the steam generators, it also reduced corrosion product/protective layer formation in the feedwater piping. This proved damaging for the feedwater piping, where flow induced erosion continued to ablate away the older corrosion/protective layer, without formation of a replacement corrosion layer, resulting in eventual pipe wall thinning and rupture at a pipe bend where flow turbulence had the greatest impact on erosion.

In view of the extensive investigation into the root-cause corrosion/erosion mechanism related to the Surry event, and the expected effect of an exacerbated erosion process at higher feedwater flow rates for uprated power conditions; it is of interest here to assess the level of review feedwater flow on piping integrity for the 1995 Surry power uprate application and review.

Licensee Application: The following are excerpts from the Surry Uprate Safety Evaluation Report with regards to assessment of the uprate conditions on feedwater performance [Ref. 3-15]:

> Licensee Submittal (see Section 4.2.4, Condensate and Feedwater Systems): The condensate and feedwater system pressures and temperatures at up-rate conditions were compared to present operating and design pressure and temperatures. There are insignificant changes to condensate and feedwater system pressure and temperature due to the uprating and these small changes are within the design capacities of the

current system. The decrease in pressure is due to condensate/feedwater pump head characteristics and increased pressure drop at increased flow rates.

The total Condensate and Feedwater System resistance was evaluated for the new flow rates pertaining to the core uprate. The steam generator pressure did not change. The overall increases are due to the greater friction losses at increased flow rates. However, it has been determined from the condensate/feedwater system calculation at uprated conditions, that the existing pumps have sufficient head to overcome the increased total system resistance with two condensate and two steam generator feed pumps in operation at the uprated condition.

The net positive suction head available (NPSHA) at the suction of the condensate and feedwater pumps was evaluated at the up rated conditions. It was determined that sufficient NPSHA exists to allow acceptable operation at the uprated flow.

4.2.5 Feedwater Heaters: Prior to the implementation of the Surry Core Uprate, the first through sixth point feedwater heaters and external drain coolers will have been replaced. The evaluation of the feedwater heaters for the up rated conditions uses the design data of the new heat exchangers. The temperatures, pressures and flow rates used in the feedwater heater review were calculated.

4.2.19 Flow-Accelerated Corrosion: The piping systems susceptible to flow-accelerated corrosion (FAC) were reviewed to assess the impact of the core uprate project. Originally FAC was called erosion/corrosion (E/C), but the industry has moved to the more appropriate term in recent years.

The thermodynamic conditions at the current power level were compared to the conditions at the projected core uprate power level, using heat balance runs provided for each condition. This comparison was utilized as the temperatures/quality and flow rates in susceptible piping systems are the key parameters for determining the core uprate impact on FAC wear rates in single and two-phase flow. The differences in operating temperatures equalities at 100 and 104.3 percent of current power levels would not tend to increase or decrease the FAC wear rates significantly. The percentage of change in flow rates is approximately proportional to the percentage of change in FAC wear rates with all the other variables affecting FAC wear rates remaining constant. This conclusion is based on available industry information and the experience of engineering in computer modeling of susceptible systems to produce a wear rate analysis.

The FAC program for Surry Power Station requires the preparation of a piping inspection list for any given refueling outage which reflects any changes in operating conditions, such as an increase in the mass flow rate. The means of determining the extent of FAC degradation is through the evaluation of ultrasonic thickness (UT) measurements of susceptible components. UT data of susceptible components will be available when each unit has experienced at least one fuel cycle at the core uprate power level. At that time, any actual increases in FAC wear will be programmatically determined and evaluated.

There are numerous other operating parameters unrelated to core uprate which can have an effect on FAC wear rates. Any predictions of future FAC wear rates must also take these parameters into account. Surry Power Station is currently evaluating increasing the pH of the secondary water chemistry. It is possible that any increase in FAC wear due to core up rate could be offset by an increase in the pH of the secondary water chemistry.

In summary, industry information and engineering experience indicates that the flow rate and temperature/quality of the media are the two factors involved in the core uprate project which could affect single or two-phase FAC. As these two parameters will not be significantly increased in the FAC susceptible systems, there should be little impact on current programmatic component wear rate predictions. The Secondary Piping and Component Inspection Program provides the tracking, trending and inspection scope for the Surry FAC effort. Therefore, any impact which does occur as a result of the core uprate project will automatically be factored into future FAC wear rate prediction models.

NRC Safety Evaluation Report: The associated NRC evaluation is limited to the following statement from SER [Ref. 3-16]:

The licensee evaluated the adequacy of the balance of plant piping systems by comparing the existing design bases parameters with the core power uprate conditions. The comparison indicated that for most piping systems, the design temperature and pressure are bounding for the power uprate. For piping systems, such as the 3rd Point Extraction Steam piping and the 5th Point Heater Drain piping, in which the temperature exceeds the design requirements of the system, the licensee performed the stress analyses for the proposed core power uprated conditions. The evaluation concluded that the 5th Point Heater Drain piping nozzle loads exceed the allowable limit. In the February 13, 1995, response to the staff's request for additional information, the licensee committed to reinforce the affected nozzle for operation at the uprated power level.

In addition, the licensee reviewed the design bases pipe break analyses to evaluate the effects of the uprate conditions on the pipe break locations, jet thrust, and jet impingement forces that were used in the plant hazard analyses and the design of pipe whip restraints. The review verified that the existing postulated pipe break locations are not affected by the power uprate since the design bases piping analyses will not change due to the power uprate. The current design bases for jet thrust and jet impingement forces due to postulated pipe breaks for these systems are not affected by the uprate since the systems do not experience a pressure increase as a result of the core power uprate. Based on its review. the staff agrees with the licensee's conclusion that the original design analyses for the pipe break locations, jet thrust, jet impingement, and pipe whip restraints are unaffected by the power uprate.

Based on the above evaluation, for all the secondary-side systems reviewed, the staff concurs with the licensee's conclusion that the power uprate has no significant impact on the balance of plant design bases.

Discussion: Flow-accelerated pipe corrosion was discussed in the licensee Uprate Safety Report, as part of the balance of plant assessment (see Section 4.2.19, Flow-Accelerated Corrosion). This evaluation included the impact of flow-accelerated corrosion on the feedwater piping at the uprated power. The primary reasoning behind the licensee conclusion that there would be nil added feedwater piping corrosion/degradation at the uprated power level was based on an estimation of differences in operating feedwater flow rates, temperatures, and coolant equality at 104.3-% (the uprated power) of the prior approved power level. A judgement was made that changes in these thermodynamic conditions would not increase (or decrease) corrosion wear rates significantly than that which have occurred at the lower prior approved power level. The licensee concluded that the percentage of change in corrosion wear rate would be approximately proportional to the percentage of change in flow rate, with the other variables affecting wear rates remaining constant. This conclusion was based on engineering models of wear rates. The NRC staff essentially concurred with the licensee's conclusion.

Root cause analysis of the event however, pointed to a more complex/synergistic process, where piping corrosion/wear involved interplay of both chemical and mechanical (flow) effects. It is believed that corrosion products on feedwater piping, which formed during earlier years of operation when untreated feedwater coolant was used, allowed for the buildup of a protective corrosion product layer on the piping walls. The use of treated water at the uprated power, however, resulted in very little corrosion product and protective layer formation. This proved damaging when in combination with the effect of the increased feedwater flow rates, which exacerbated turbulence at piping bend locations where the break occurred. The higher flow rates and turbulence apparently caused ablation of the previously formed protective layer, thus resulting in pipe wall thinning and ultimate pipe rupture. Although chemical and flow effects were considered independently in the licensee's uprate submittal, consideration of the combined or synergistic interplay between chemical and mechanical (flow) effects was not evident from review of the applicant's submittal or the NRC-SER review of that submittal.

The Surry-2 pipe rupture event is just one of a number of rather dramatic pipe rupture events that continue to plague the industry. More recent examples include the Aug. 11/99 event at the Callaway-1 plant (power uprate in 1988), where a double-ended guillotine break occurred in an 8" diameter steam line leading to a feedwater heater; and the Japanese Tsuruga-2 event of July 12/99 (operating at an electrical power output greater than specified for the original design) where a large-leak crack developed in the primary piping at an elbow location in the letdown line from the regenerative heat exchanger. A compilation of pipe ruptures has been recently documented in an EPRI report [Ref. 3-17], indicating in excess of 170 dramatic pipe rupture events in LWRs, ranging from single-ended pipe breaks to full double-ended guillotine ruptures of the Callaway type. The cause of such ruptures is generally due to flow/erosion or flow-assisted corrosion effects for pipes greater than 1" in diameter. Flow-assisted ruptures would be expected to be exacerbated at the higher flow rates that generally accompany a power uprate, therefore, piping rupture potential deserves considerable attention in power uprate reviews. Besides dramatic ruptures, the EPRI compilation [Ref. 3-17] presents piping failure data for a broad range of breaks, ranging from non-leaking cracks to large double-ended guillotine failures resulting which have resulted in plant staff fatalities. In excess of 4000 piping breaks are noted in the EPRI report.

3.4 Brunswick Power Uprate Review

Event Description: A power uprate amendment was granted for Brunswick 1 & 2 (BWR) units on Nov. 1, 1996, which allowed for an increase in power from 2436 to 2558 MWt for each unit. During the course of the power uprate application the licensee and NRC staff believed that the design temperature limit for the wet well torus to be 220°F. On November 4, 1996, the licensee reported that plant technical specifications listed a more restrictive value of 200°F for the torus design temperature [Ref. 3-18]. Licensee analysis for design basis accidents (DBA) indicated peak wet-well coolant temperature of 201°F could be reached at the uprated power level, which would exceed the 200°F design limit. Of note, is the fact that the discrepancy was found by the licensee rather than the NRC staff performing the uprate review.

Licensee Application: The following table, which summarizes Containment Performance Results, is abstracted from the utility's Uprate Licensing Report [Ref. 3-19, Table 4-1], showing the presumed torus design temperature of 220°F (Peak Bulk Pool Temperature Limit).

Table 4-1. Containment Performance Results

Parameter	Current Rated Power	Uprated Power	Limit
Peak Drywell Pressure (psig)	49.4 (UFSAR) 36.8 (LTP Method)	38.1 (LTP Method)	62
Peak Bulk Pool Temperature (°F)	205 (UFSAR) 197 (current method)	201 (current method)	220 used in uprate analysis 200 is correct value
Drywell Temperature (°F)	283	284	340

NRC Safety Evaluation Report: The associated NRC SER [Ref. 3-20] does not mention any check or verification of the validity or correctness of information in the above table.

Discussion: Although this event does not indicate concerns related to synergistic effects, it does point to a less than thorough assessment by the NRC staff of the correctness of information submitted by the licensee. It is of particular note, since it involves not only a lack of an independent check of the validity of DBA analysis by the NRC staff, where a re-evaluation of DBA conditions at the uprated power is considered a key element of uprate application/review process. The event also indicates a degree of unfamiliarity of the NRC review team with plant specific design limits.

3.5 Limerick Power Oscillation Event

Event Description: In March 1994, during startup of Limerick-1 from a refueling outage, core criticality was achieved earlier than expected, with code predicted delta-K/K less than that measured with in-core detectors. The plant operator determined that the predicted shutdown margin was non-conservative by 0.78-% delta-K/K [Ref. 3-21]. The event illustrates a reoccurring problem noted for high-power cores utilizing high-burnup fuel assemblies, where core physics modeling capability has not been adequately updated to account for the complexities of core reload schemes incorporating an array of fresh, medium, and high-burnup fuel assemblies of various vendor designs, and which are operated at considerably higher power densities than that for which the code were originally benchmarked.

The Limerick-1 incident involved fuel bundles of the highest enrichment and highest gadolinium concentration ever loaded at Limerick. Prior to startup the utility compared core physics predictions from a code supplied by the fuel vendor (vendor code) and another from an engineering support firm (alternate code). The alternate code included enhancements to more accurately model the steeper neutron flux gradients characteristic of high power density cores. Although the vendor and alternate cores produce notable differences in reactivity predictions, procedures did not require reconciliation of prediction differences. An investigation after the event indicated that the alternate code more accurately predicted core performance than the fuel vendor code, although it was the fuel vendor predictions that were the basis for the reload analysis used by the plant operator.

Licensee Application: Although the 1994 delta-K/K event occurred prior to the 1996 power uprate approval, submittal information for the uprate request [Ref. 3-22] with regards to core neutronics and fuel performance predictive capabilities are of interest.

> *2.1 Fuel Design and Operation*: At original or re-rated conditions, all fuel and core design limits will continue to be met by control rod pattern and/or core flow adjustments. New fuel designs are not needed for power rerate to assure adequate safety. However, new fuel enrichments may be used to provide additional operating flexibility and maintain fuel cycle length.
>
> The reactor core design power distribution represents the most limiting thermal operating state at design conditions. It includes allowances for the combined effects on the fuel heat flux and temperature of the gross and local power density distributions, control rod pattern, and reactor power level adjustments during plant operation. Thermal-hydraulic design and operating limits assure an acceptably low probability of boiling transition-induced fuel cladding failure occurring in the core at any time, even for the most severe postulated operational transients. Limits are also placed on fuel linear heat generation rates in order to meet both peak cladding temperature limits for the limiting Loss-of-Coolant Accident (LOCA) and fuel mechanical design bases.
>
> The subsequent reload core designs for operation at the rerated power level will take into account the above limits to assure acceptable differences between the licensing limits and their corresponding operating values. Power re-rate will increase the cores' average power density. However, this power density is still well within the current operating power density range of other BWRs. The power rerate will have some effects on operating flexibility, reactivity characteristics, and energy requirements. These issues are discussed in the following sections based on GE experience with power rerate and fuel characteristics.
>
> *2.3 Reactivity Characteristics*: All minimum shutdown margin requirements apply to cold conditions, and will be maintained without change. Operation at higher power could reduce the excess reactivity (typically by about 0.2 to 0.3 % delta-K for a 5-% power increase) during the cycle. This loss of reactivity is not expected to degrade the ability to

manage the power distribution through the cycle to achieve the target power level. However, the lower reactivity does result in achieving an earlier all-rods-out condition. Through fuel cycle redesign, sufficient excess reactivity can be obtained to match the desired cycle length. The increase in hot reactivity may result in less hot-to-cold reactivity difference and, therefore, smaller cold shutdown margins. However, this loss in margin can be accommodated through core design. If needed, a bundle design with improved shutdown margin characteristics can be used to preserve the flexibility between hot and cold reactivity requirements for future cycles.

NRC Safety Evaluation Report: The NRC granted a power uprate request for Limerick-Unit 1 on January 24, 1996, although cognizant of the 1994 delta-K/K problems. The following statements are taken from agency's safety evaluation report [Ref. 3-23] for the uprate, related to evaluation of the utilities analysis of core performance:

> *3.1 Reactor Core and Fuel Performance*: The staff evaluated the power uprate for its effect on areas related to reactor thermalhydraulic and neutronic performance such as the power/flow operating map, core stability. reactivity control, fuel design, control rod drives, and scram performance. The staff also considered the effect of power uprate on reactor transients, anticipated transients without scram (ATWS), ECCS performance, and peak cladding temperature for design basis accident break spectra.
>
> *3.1.2 Power/Flow Operating Map*: The uprated power/flow operating map includes the operating domain changes for uprated power. Changes to the power/flow operating map are consistent with previously approved generic descriptions. The maximum thermal operating power and maximum core flow correspond to the uprated power and the previously analyzed core flow range. Uprated power has been re-scaled, so that it is equal to 100-% rated.
>
> *3.1.3 Stability*: The licensee evaluated the effect of power uprate on core stability issues according to the generic guidelines for power uprates. To determine the effect on core stability, the licensee reviewed recommendations from GE Service Information Letter (SIL-380, Rev.1), NRC Bulletin 88-07, and current Boiling Water Reactor [BWR] Owners Group (BWROG) efforts, including interim corrective actions (ICAs) recommended by GE and the BWROG. The licensee adjusted the percent power on the revised power/flow map such that the ICA region boundaries have the same actual power (MWt); thus Units 1 and 2 will have the same level of protection against thermal-hydraulic instability. Furthermore, the analysis shows that the power increase will not affect the application of any of the BWROG stability long-term solution options.
>
> The staff concludes that the licensee addressed thermal hydraulic stability in an acceptable manner.

Discussion: In addition to Limerick, other events have likewise involved inadequate or faulty reload analysis [see Ref. 3-24]. For example, a core physics verification test for the Duane Arnold plant in April 1995, indicated a shutdown margin of 0.33-% delta-K/K, while plant technical specifications called for a minimum margin of 0.38-% delta-K/K . Similar to Limerick, core physics software did not accurately predict reactivity for the reload design. Specifically, an insufficient number of data points were used to model reactivity changes from burnable poison depletion at elevated burnup levels. Another example includes WNP-2, where criticality was achieved earlier than predicted, due to an error in the xenon decay model for extended burnup levels. Each of these events involved discrepancies in predicted versus measured reactivities for core reloads involving new fuel designs or longer fuel cycle lengths, using core physics codes which had not been validated at the relatively high power densities for which they were being applied. A more through agency assessment of the adequacy of core physics models at elevated power levels, which would appear to be specifically warranted during the power uprate review process, is indicated by the events involving faulty reload analysis cited above.

3.6 REFERENCES

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4. POWER UPRATES AND SYNERGISTIC EFFECTS

The impact of elevated core densities, when combined with potential system and component degradation via plant aging processes and fuel life extension are considered in this chapter. It is cautioned however, that little direct information is available on the subject; thus this discussion is somewhat conjectural in nature and primarily serves as a cautionary note of potential concerns for aged/long-fuel cycle population of reactors.

4.1 Plant Characteristics Impacted by Power Uprates

An uprate evaluation generally begins with an assessment of primary and secondary thermal hydraulic conditions required to achieve the uprated power level for the plant. Since ultimate power level is generally dictated by turbine-generator capacity, the actual range of plant thermal-hydraulic conditions is somewhat limited. The analysis centers on reactor and steam generator heat balances, which define the range of primary and secondary coolant flow and temperature conditions for the uprated power. Key parameters which may be altered by a power increase include:

Primary Coolant System (PCS)	Secondary Coolant System (SCS)
Core Power	Steam Generator Steam Flow Rate
Core Inlet/Outlet Enthalpy	Feedwater Flow Rate
Vessel Outlet/Inlet Coolant Temperatures	Feedwater Temperature
Fuel Temperature	Feedwater Pumping Requirements
Primary Coolant Flow Rate	
Primary Coolant Pumping Requirements	

Tables 4-1 and 4-2 present changes in such thermal-hydraulic parameters for typical PWR and BWR uprate conditions. Synergistic effects related to power uprates thus largely stem from changes in such thermal-hydraulic conditions, in combination with some other physical process. Examples include increased coolant flow effects in combination with corroded/aged piping, or increased fuel temperatures in conjunction with elevated burnup conditions. Such synergistic effects are considered in this chapter.

4.2 Potential Synergistic Power Uprate/Fuel Burnup Effects

Any core power increase will result in a corresponding increase in the total coolant enthalpy rise across the core. Typically an increase in core enthalpy is achieved through the an increase in total coolant flow in conjunction with a small increase in core coolant temperatures. In some situations however, the enthalpy gain is achieved solely through an increase in coolant flow, without any increase in the coolant temperature conditions from that at the prior power level. Since power uprates do not generally involve significant increases in coolant or fuel temperatures, the discussion here centers on flow related effects in combination with longer fuel duty times associated with extended-life/high-burnup effects

Table 4-1. Prior and Uprated Conditions for Typical BWR Plants

	Monticello Plant		Brunswick Plant	
Parameter	Prior Value	Up-rated Value	Prior Value	Up-rated Value
Thermal Power (Mwt)	1670	1775	2436	2558
(% Power Uprate)	—	6.28-%	—	5.0-%
Vessel SteamFlow (lb/hr)	6.78	7.26	10.47	11.077
Full Power Core Flow Rate (lb/hr)	43.2-60.5	47.4-60.5	57.8-80.8	62.4-80.3
(% Rated Core Flow)	(75-105)	(82-105)	(75-105)	(81-104.3)
Nominal Operating Dome Pressure (psia)	1025	1025	1020	1045
Dome Temperature (F)	548	548	547	550
Turbine Inlet Pressure (psia)	965	965	965	1000
Full Power Feedwater				
Flow (lb/hr)	6.75	7.23	10.44	11.054
Temperature (F)	377	383	420	425
Core Inlet Enthalpy (Btu/lb)	524.6	523.7	526.9	529.7

Table 4-2. Prior and Uprated Conditions for the Surry PWR Plant

	Prior Value	Uprated Value
Reactor Pressure Vessel Conditions		
Reactor Power, MWt	2441	2546
Percent Power Increase, %	---	4.3
Reactor Coolant Pressure, psia	2250	2250
Total Reactor Flow, lb/hr	100.7x10E6	101.1x10E6
Thermal Design Flow, gpm per loop	88,500	88,500
Reactor Coolant Temperatures, F		
Vessel/Core Inlet	543.0	540.4
Core Outlet	608.2	609.3
Vessel Outlet	605.6	605.6
Steam Generator Conditions		
Steam Temperature, F	516.0	515.9
Steam Generator Outlet Temp, F	543.0	540.1
Steam Pressure, psia	785	784
Total Steam Flow, lb/hr	10.66x10E6	11.26x10E6
Feedwater Temperature, F	437.7	443.2

During the 1970-80s lead fuel rod demonstration programs [Refs. 4-1 through 4-4] involved successful irradiation of several thousand rods to burnups approaching 60,000 MWD/t, where good overall fuel performance was demonstrated. However, the test rods in these demonstration programs were generally irradiated at non-aggressive core locations, that is at locations which did not involve peak fuel or coolant temperatures, peak neutron fluence, or high coolant flow conditions. Little direct information can therefore be extracted from these tests for extended fuel life/uprated-power conditions. The discussion here is thus based on trend observations, and centers on Zircaloy cladding water-side corrosion, fuel rod fretting and fuel rod bow/buckling effects.

4.2.1 Zircaloy Cladding Oxidation/Corrosion: Zircaloy quickly oxidizes at room temperatures in both air and water environments, so that typically fresh Zircaloy cladding has a thin adherent oxide layer at the surface which protects it from further rapid oxidation. For an extended fuel cycle, accumulated cladding oxidation depends on in-reactor residence time, coolant temperatures, and water chemistry. Experience from the lead-fuel/high-burnup demonstration programs consistently shows slow but continued water-side corrosion of Zircaloy cladding with increased in-reactor residence time (burnup). Such water-side corrosion is typically exacerbated by crud buildup on the rod surface, which tends to retard rod-to-coolant heat transfer, thereby resulting in local hot spot formation and an attendant increase in local Zircaloy oxidation/corrosion. The root cause of crud formation is precipitation of inorganic solids from coolant impurities; thus, adequate control of water chemistry is essential for prevention of adverse crud buildup and rod oxidation for extended duty times. The effect of an increase in coolant flow stemming from uprated power conditions in itself may not increase crud formation potential or Zircaloy water-side corrosion; nevertheless, Zircaloy oxidation would increase with in-reactor residence time.

4.2.2 Hydrogen Uptake: Zircaloy cladding absorbs a portion of the free hydrogen generated during corrosion, which can alter Zircaloy mechanical properties, principally ductility. At reactor operating temperatures absorbed hydrogen remains in solution within the alloy, up to about 200 ppm, at which point Zirconium-hydride precipitates form. Such precipitates do not significantly alter the ductility of Zircaloy, so long as the concentration remains below 400-500 ppm. The ductility of hydrided Zircaloy is altered however during cool-down. At low temperatures hydride platelets form, which preferentially orient parallel to the basal planes of the hexagonal zirconium structure resulting in a more rigid crystalline structure. To alleviate loss of ductility at low temperatures, Zircaloy cladding is generally fabricated to produce a crystallographic texture that favors circumferential hydride platelet orientation.

Results from high-burnup fuel demonstration programs indicate that there is a continued increase in hydride formation with increased burnup, with an attendant loss of cladding ductility. Although hydriding has not lead to premature cladding failures for operational conditions, such hydriding and loss of ductility is believed to be largely responsible for high burnup fuel failures noted in recent RIA (Reactivity Insertion Accidents) tests [Refs. 4-5 through 4-7]. Hydriding effects would also be expected to cause early cladding failures for loss-of-coolant accidents (LOCAs). Since Zircaloy guide tubes and grid spacers are relatively thin structures and are subjected to two-sided exposure to coolant and associated hydrogen uptake, hydriding/embrittlement effects also impact Zircaloy based guide tubes and grid spacer performance.

For somewhat higher primary coolant flow rates associated with uprated power conditions no significant impact on hydrogen uptake would be expected. Thus synergistic effects of higher coolant flow rates in combination with longer fuel duty times would not be anticipated, where loss of cladding ductility largely stems from extended fuel duty times.

4.2.3 Fuel Rod Bowing and Fretting: Fuel rod bowing is traditionally considered a licensing issue because coolant channel reduction can lead to Departure from Nucleate Boiling (DNB). So also is fuel rod axial growth, since it can lead to rod buckling/bowing. The lead fuel demonstration programs of the 1970-80s showed an approximate linear trend of increased axial growth with increased fuel duty times. Recent experience with control rod sticking problems [Refs. 4-8 through 4-10] for certain Westinghouse fuel assemblies indicate irradiation induced axial growth of the guides tubes through which the control rods are inserted into the core. The problem was first noticed at South Texas-1 plant on Dec. 1995, when three control rods failed to fully insert when the unit was tripped from 100-% power. The control rods in question involved a Westinghouse XLR type fuel assembly at a burnup level of ≈ 42,880 MWD/t. Subsequently five control rods at Wolf Creek failed to properly insert during core trip, involving Westinghouse 5H fuel assemblies irradiated to 47,6000 MWD/t. Control rod problems were also noted at the North Anna-I plant in Feb. 1996, during retrieval of new control rods temporarily stored in the plant's spent fuel cooling pool within high-burnup fuel assemblies (47,780-49,600 MWD/t-U). The root cause of such insertion problems has been tied to distortion of the Zircaloy tubes within which the control rods are inserted into the core. Guide tube distortion apparently resulted from accelerated axial growth of the Zircaloy tube in conjunction with high local temperatures. Such irradiation induced guide tube axial growth lead to guide tube distortions that prevented full control rod insertion.

An associated concern related to fuel life extension is *rod fretting*, that is Zircaloy cladding mechanical wear caused by flow associated rod vibration and contact wear with adjacent structures (grid spacers, adjacent fuel rods, adjacent thimble tubes, etc.). In general, intentional grid-to-rod contact force is incorporated into fuel assembly designs to prevent rod fretting. Synergistic effects of higher coolant flow rates in combination with longer fuel duty times may however, lead to rod fretting. On one hand, irradiation associated Zircaloy growth for extended fuel cycle times clearly has the potential to cause Zircaloy axial expansion and distortion, as recently demonstrated by control rod thimble tube distortion and attendant control rod insertion problems. Likewise, uprate applications requiring higher coolant flow rates at local core regions could induce rod vibration at such high-flow locations. The combination of higher coolant flow rates (increase vibration potential) in concert with longer fuel duty times (increased rod distortion potential) could thus lead to a deleterious rod fretting process, which should be considered in power uprate evaluations.

4.2.4 Boron Effects: Reactivity control in PWRs is generally augmented by the addition of boric acid to primary coolant, where boron-10 has a high cross-section for neutron capture. High boron concentrations are also used at the beginning of a new power cycle to compensate for excess reactivity for long-cycle/high-burnup cores. Several incidents of unanticipated core performance have been noted for highly borated cores. Axial power offset is one such problem [Ref. 4-11], which appears tied to crud buildup and the gettering of boron by crud. Such crud-buildup/boron-gettering effects are generally found at the upper elevations of high burnup fuel assemblies where the effect is exacerbated. The crud appears to getter boron, causing a distortion of the axial power profile. Crud buildup appears to go hand-in-hand with longer fuel duty times and since boron gettering by such crud is evident, reactivity perturbations can be expected for elevated burnup conditions, independent of power level. However, there also appears to be evidence for synergistic implications, where the effect is compounded at high-power core locations.

High boron concentrations are also thought to have a deleterious effect on cladding corrosion, where observations indicate boron being the culprit for enhanced Zircaloy cladding corrosion noted at Crystal River-3 and TMI-1 reload cores. Such boron-associated corrosion impacted all fuel rods, including those with only 115 effective full power days [Ref. 4-11, 4-12]. The exact cause for such boron assisted corrosion is not known at this time.

An assessment of the full/synergistic implications of high boron concentrations and crud formation, would appear to be warranted as part of any future power uprate application and review by the agency.

4.3 Potential Synergistic Power-uprate/Aging Effects

During the 1980s and early 1990s the U.S. Nuclear Regulatory Commission conducted a comprehensive, hardware-oriented, research program to understand ageing mechanisms of components and systems in nuclear power plants [Refs. 4-13 through 4-15]. The main feed-water pipe break at Surry-2 in 1986, which resulted in four site-worker fatalities, offers a stark example of the safety implications of ageing. Post-accident investigations revealed pipe failure due to corrosion/erosion effects, one form of ageing. Other incidents of age associated degradation of nuclear plant components include steam-generator tube degradation, electric cable embrittlement by heat/radiation damage, and safety/relief valve wear.

Ageing of reactor components and systems results from long-term exposure to a rather harsh environment of steam, corrosive chemicals, and mechanical wear and vibration, exacerbated by radiation and high-temperatures. Virtually every component in a nuclear power plant is subject to some form of ageing, from the fuel rods (corrosion) and reactor pressure vessel (irradiation embrittlement), to coolant pipes (corrosion/erosion), pumps and valves (wear), and electrical (insulation degradation) systems. The main regulatory concern is that plant safety could be compromised if degradation of key components and systems is not detected, or if the ability to take timely corrective action is impaired. Adequate understanding of ageing effects is of importance when considering plant life extension and license renewal, particularly in a very cost-competitive/deregulated environment, where plant efficiency and cost recovery arguments would dictate reactor operation at high power densities. The synergistic impact of aging effects in combination with high power densities (uprated power levels) are considered here. The discussion centers on reactor system components addressed in NRC's aging research program highlighted in Table 4-3.

TABLE 4-3. Listing of NRC Sponsored Aging Research

- Ageing Assessment of Electrical Components Batteries, battery-chargers & invertors, circuit-breakers & relays, electrical cables & penetrations, motors, connectors, switches, resistance temperature detectors, transformers, surge arrestors, diesel generators
·Ageing Assessment of Fluid-Mechanical Components P:ipes, pumps, motor-operated valves, check valves, power-operated relief valves, safety relief valves, main steam isolation valves, snubbers, heat exchangers, compressors, fans, air-operated valves.
- Ageing Assessment of Instrumentation and Control Systems Reactor protection system, class 1E distribution systems, motor control centers, control rod drives
- PRA and Modeling Efforts Data and record-keeping treatment of ageing in passive structures, PRA evaluation of ageing phenomena, residual life evaluation of major LWR components, degradation modeling

4.3.1 Ageing of Electrical Components: Nuclear power plant electrical components include batteries, battery chargers and inverters for essential equipment, circuit-breakers, relays and switches, electrical cables and penetrations, motors and diesel generators, as well as items such as resistance temperature detectors, to name a few.

As discussed in the GALL report [Ref. 4-13], age-related cable degradation is primarily due to radiation and elevated temperature induced degradation of the cable insulation and jacket. Despite considerable NRC research to develop techniques for detection of insulation degradation, a quantitative determination of the extent of insulation breakdown is difficult. Although electric parameters such as resistivity are relatively easy to measure, they are not easily correlated to insulation integrity or degradation.

NRC research efforts related to electrical circuit-breakers and relays on essential equipment, show aging/failure mechanisms related to contact wear and loss of lubrication, primarily associated with elevated temperature environments. Energized relay coils tend to exhibit high failure-rates because of their insulation breakdown is caused by self-heating from the continuous current in the energized coil. However, electric circuit breakers and relays on essential equipment are routinely refurbished on a periodic schedule and covered under the Maintenance Rule. Likewise, items such as batteries, battery chargers and inverters can be taken out of service for maintenance because of their redundancy [Ref. 4-16]. Such electrical components are also included in plant quality assurance programs that require periodic replacement.

The primary concern related to aging of electrical components therefore centers on cable degradation and the associated lack of an adequate warning for cable insulation breakdown. Since insulation breakdown appears to be a cumulative effect, with increased degradation due to cumulative radiation exposure levels exacerbated by elevated temperatures, synergistic degradation can be expected for aged plants in conjunction with power uprate conditions. Potential compounding effects of plant life extension in combination with elevated power conditions should be considered.

4.3.2 Ageing of Fluid-Mechanical Components: An extensive array of fluid-mechanical components form an integral part of the primary and secondary cooling systems of any nuclear power plant. Such mechanical components include pumps, piping, and various types of valves (motor-operated valves, check valves, power-operated relief and air-operated valves, safety relief valves), as well as snubbers, heat exchangers, to name a few.

Corrosion and corrosion-related processes are the dominant mechanisms of age-related degradation of coolant piping [Refs. 4-17] . NRC research has shown that pipe erosion/corrosion is exacerbated at increased fluid velocities. Additionally, non-uniform water temperature fields aggravated by thermal buoyancy can cause large induced structural thermal stresses, which can lead to cracking or significant structural distortion of coolant pipes. These thermal stresses are usually not accounted for in component design and are highly plant- and mode-of- operation dependent. They can occur under normal or intermittent operation of plant systems and tend to be worse under low flow conditions. For reactor internals, irradiation-assisted stress corrosion cracking is also a source of pipe degradation where high radiation fields are present. Other forms of corrosion, as well as vibrational fatigue, can contribute to degradation. The higher coolant velocities associated with power uprated conditions would therefore be expected to aggravate piping aging. The Surry event is an example of age related piping corrosion, which was apparently exacerbated by the higher coolant flow rates stemming from uprated power conditions. The full implications of such synergistic effects should be evaluated in future power up-rate/license extension reviews by the agency.

Pump and valve casings were likewise found to be subject to corrosion and erosion /corrosion related degradation. Thermal embrittlement was an important mechanism in cast stainless steel pump and valve components. Moving parts in pumps and valves suffer from age related degradation produced by wear, vibration, fatigue, and erosion /corrosion. Elastomer components, such as valve and pump seals, are also subject to degradation by physical and chemical attack, particularly at elevated temperatures. Consideration during power uprate reviews, of the potential for compounded degradation due to prolonged component life (aging) in conjunction with the increased coolant flow and temperatures for uprated conditions, is recommended.

4.3.3 Ageing of Instrumentation and Control Systems (I&C): The reactor protection system, various in-core instrumentation, motor control units, and control rod drive systems generally comprise systems classified as instrumentation and control systems. Instrumentation and control systems are made up of many small components that are routinely replaced after a number of years of service, as determined by qualification programs. Thus, I&C aging is largely controlled by scheduled maintenance and periodic replacement of components of most I&C systems. Redundancy of many components that comprise the reactor protection system and engineered safety system, allows for replacement of such components as part of plant maintenance schedules [Ref. 4-18]. Motors and generators for the reactor protection and engineered safety systems are known to fail due to bearing wear caused by vibration and winding insulation breakdown from elevated temperatures. Motor brushes also fail due to wear. Since synergistic aging/power level effects can be envisioned for components of I&C systems, due to prolonged component life (aging) in conjunction with the increased coolant flow and temperatures associated with elevated power levels, consideration of such synergistic effects should be included in I & C component replacement evaluations for aged plants operating at elevated power levels.

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5. SUMMARY and CONCLUSIONS

Since the early 1980s, the NRC has reviewed and approved several dozen power uprate requests by licensees. Additional applications to operate at even higher power levels are expected for the deregulated/cost-competitive utility environment of the future. Although the NRC staff reviews all power increase requests to assure that regulatory requirements are met, there is a concern that potential synergistic effects may not be adequately considered in such reviews; namely that high core power densities when combined with aging associated system/component degradation and fuel-life extensions to high burnup (>50 GWD/t-U) may adversely impact safety. This investigation thus centers on an assessment of agency power uprate review practices and evaluates the need to consider potential synergistic effects[a] .

The investigation of agency review practices for power uprate applications focused on a review of documentation for seven representative power uprate applications, namely that for Hatch-BWR, Limerick-BWR, Maine Yankee-PWR, Monticello-BWR, North Anna-PWR, Surry-PWR, and Wolf Creek-PWR plants. For each plant the documents reviewed included the Uprate Amendment Licensing Report submitted by the Licensee, and the associated Safety Evaluation Report (SER) which documents findings from the NRC staff review of uprate request.

The Uprate Amendment Licensing Report largely centers on the licensee's analysis and re-evaluation of Design Basis transients and accidents at the uprated power level. Changes to plant technical specification for operation at the increased power level are also identified and justified by the licensee. The NRC staff reviews the analysis submitted by the licensee for accuracy, and verifies compliance at the increased power level with criteria and conditions specified in the plant's original Final Safety Analysis Report (FSAR). The staff also makes an evaluation of any potential unreviewed safety questions that might occur as a result of the increased power rating in accordance with 10CFR-Part 50.59. The uprate application is generally approved if the case has been made that the plant can be operated at the uprate power level without a significant increases in the amount of effluents or radiation emitted from the plant, a nil reduction in safety margins for both operational transients and Design Basis Accidents (DBA), and that no new or different accidents would occur at the increased power than those considered in the original Safety Analysis Report (FSAR) and license basis.

Results of this investigation revealed essentially nil documentation with regards to either licensee consideration of potential synergistic effects or agency requests for information on potential synergisms. Concerns regarding the potential for diminished safety margins due to the combined impact of uprated power levels in conjunction with system/component degradation via plant aging and/or fuel-life extensions to high burnup (>60 GWD/t-U), were not evident from the review of the uprate documentation for the seven plants surveyed. A review of operational events and other incidents however shows some evidence, albeit indirect, that synergistic effects may indeed occur.

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a. *Synergistic: the cooperative action of discrete agencies such that the total effect is greater than the sum of the effects taken independently.*

The main feedwater pipe break at Surry-2 in 1986, resulting in four site-worker fatalities due to release of scalding steam, provides an example of potential synergistic aging/flow effects. Post-accident investigations revealed pipe wall thinning and rupture due to corrosion (one form of aging), which appeared to be exacerbated by a feedwater flow erosion effect (higher feedwater flow rates are characteristic of uprated power conditions).

Other examples include the control rod insertion problems which occurred at the Wolf Creek and North Anna PWR plants, both having received prior power uprate approvals in the range of 4 to 5 percent. At the Wolf Creek plant, five control rods failed to fully insert during scram from full power. The affected control rods involved Westinghouse vantage 5H fuel assemblies with burnups greater than 47,6000 MWD/t-U. Root cause analysis indicated Zircaloy thimble tube distortion (tubes within which the control rods are inserted) due to irradiation induced growth and distortion of the thimble tubes (the tubes are fixed to support plates, thus axial growth results in compressive loading of the tubes and distortion). Since irradiation damage is both a fluence dependence (i.e. power level) and time dependence (burnup level) process, synergistic effects are evident.

Control rod problems likewise occurred at the North Anna-1, where retrieval difficulty was noted for two *new* control assemblies, temporarily stored in the plant's spent fuel pool. The two affected control assemblies were stored in spent Westinghouse Vantage-5H fuel assemblies, which had achieved burnups of 47,700 MWD/t-U and 49,600 MWD/t-U. The licensee determined that the cause of the binding was related to fuel rods in assemblies with high burnup and not the control assemblies themselves. Control rod drag tests indicated a correlation of increased control rod drag force with increased fuel burnup, with a dramatic increase in drag force at fuel burnups greater than 45 GWD/t-U. An examination of both the Wolf Creek and North Anna Uprate Licensing Reports, as well as the NRC Safety Evaluation Reports for these uprate requests, did not reveal any assessment of synergistic consequences of fuel or control rod operational conditions or margins, due to the combined effects of higher power and the elevated fuel burnup conditions.

It is again noted that agency investigations into the Maine Yankee allegations corroborated whistle blower contentions of a deliberate submittal of faulty LOCA analysis provided by the licensee, as well as faulty applicant estimates for containment pressures at the uprated power level. The salient point is that such inadequacies were not revealed by NRC staff during the initial uprate review, indicating a less than satisfactory power uprate review and approval process. Both the Maine Yankee and Brunswick incidents indicate to this reviewer the need for agency capabilities to perform independent/in-house thermal-hydraulic and neutronic code analysis, to verify the accuracy of licensee submittal analysis. Such in-house computational efforts are not evident from review of the SERs examined in this study. Agency computational capabilities would go a long way in providing an independent check and verification of what is now essentially a licensee effort. In-house computational capabilities would also facilitate a better staff understanding of the nuances of computational methods and how they can be manipulated to provide desired results, as occurred in the Maine Yankee submittal.

In addition to evidence from plant events pointing to synergistic effects, a review was also made of specific parameters impacted by power uprates, when combined with plant aging and fuel life extensions. As discussed in Section 4, a core power uprate is characterized by an increase in the total coolant enthalpy rise across the core, where an increase in core enthalpy is often achieved through an increase in total coolant flow. Higher coolant flow rates, in combination with longer fuel duty times, can be expected to impact rod mechanical wear (fretting) conditions. On one hand irradiation induced Zircaloy axial expansion due to a longer fuel cycle (burnup) tends to increase the potential for rod distortion (note control rod insertion problems). On the other hand, higher coolant flow rates associated with uprated power conditions tend to increase rod vibration potential, with an associated increase in the likelihood of rod fretting/mechanical wear. The combination of higher coolant flow rates in concert with longer fuel duty times could lead to a deleterious rod-distortion/fretting process.

Another potential synergistic process is that associated with the high boron concentrations (in the form of boric acid added to reactor coolant) typically used in PWRs to compensate for excess reactivity for long-cycle/high-burnup cores. Several incidents of unanticipated core performance have been noted for long-cycle/high power density cores, including axial power offset tied to crud buildup on long-life/high-burnup fuel rods. Crud-buildup has been noted at the upper elevations of such high burnup fuel assemblies. The crud appears to getter boron, causing a distortion of the axial power profile. There also appears to be evidence for synergistic implications, where the effect is compounded at high-power core locations.

Aged reactor components and systems, in combination with high core power densities, may likewise produce degradation that is greater than the sum of the individual effects. Age degradation of reactor components results from long-term exposure to a rather harsh environment of steam, corrosive chemicals, and mechanical wear and vibration, exacerbated by radiation and high-temperatures. Virtually every component in a nuclear power plant is subject to some form of ageing, from the fuel rods (corrosion) and reactor pressure vessel (irradiation embrittlement), to coolant pipes (corrosion/erosion), pumps and valves (wear), and electrical (insulation degradation) systems. A primary concern related to aging of electrical components centers on cable degradation and the associated lack of an adequate warning for cable insulation breakdown. Since insulation breakdown appears to be a cumulative effect, with increased degradation due to cumulative radiation exposure levels exacerbated by elevated temperatures, synergistic effects may occur for aged cables subjected to higher environment temperatures characteristic associated of uprated power conditions.

Corrosion/erosion processes are a principal concern for aging of fluid-mechanical components, such as piping and valves. NRC research has shown that pipe erosion/corrosion is exacerbated at increased fluid velocities. Additionally, non-uniform coolant temperatures aggravated by thermal buoyancy can cause piping thermal stresses, which can lead to cracking and pipe distortion. Another aging factor for fluid-mechanical components is vibrational fatigue, particularly for pumps and valves in the vicinity of moving machinery. The higher coolant velocities associated with power uprate conditions would therefore be expected to aggravate piping aging. The Surry event offers an example of age related piping corrosion, exacerbated by the flow erosion. For reactor internals, irradiation-assisted stress corrosion cracking is an additional source of pipe degradation. Valve seals and other elastomer components are also subject to physical/chemical degradation at elevated temperatures and for prolonged duty times.

The potential for compounded degradation due to prolonged component life (aging) in conjunction with the increased coolant flow and temperatures associated with elevated power levels has not been adequately addressed in power uprate applications.

In view of such observations the following recommendations are made:

- Prior recommendations of the Maine Yankee Lessons Learned Report for formulation and implementation of a **Standard Review Plan** for Power Uprates are supported by observations noted in this study. Efforts in this regard should be expedited.

- A Standard Review Plan for Power Uprates, when adopted, should include acceptance criteria which consider the influence of **synergistic effects,** specifically high fuel burnup levels and component/system aging effects in combination with uprated power conditions.

- NRC staff review procedures for power uprate applications should include requirements for independent staff analysis (i.e. **thermal-hydraulic and neutronic code predictions**) and verification of uprate plant predictions submitted by the licensee. Such staff audit calculations should be included in the SER for each uprate application, including comparisons with licensee analysis.

- Since formulation/adoption of a Standard Review Plan for Power Uprates is a timely process, during the interim the NRC should establish guidelines for a more **standardized format for documentation of licensee uprate evaluations**. Standardization would allow for improved NRC staff assessment of the impact of a power uprate on plant safety margins and the generic implications of power uprates. Standardization of licensee predictions should also foster better plant-to-plant uprate comparisons. A standardized format should include:

  A) Standardized scope/content requirements should be adopted for power uprate License Amendment Reports.

  B) Standardized requirements should include specification of comparative tables (or some other format) of code input parameters and assumptions used in the License Amendment Report, at the uprated and FSAR power levels.

  C) Comparative tables should also be required, at the uprated and FSAR power levels, for code-predicted plant thermal-hydraulic and core physics conditions for operational, off-normal/transient, and design basis accident conditions.

  D) A comparison of safety measures (e.g. CDF, QHO, LERF) at the uprated and FSAR power levels, utilizing plant-specific PSA (Probabilistic Safety Analysis) results should a included in uprate applications.

APPENDIX A

Wolf Creek Uprate License and NRC Safety Evaluation Reports

This appendix presents several examples of the type information and depth of analysis typical of what is found in an up-rate request. The information provided here is abstracted *verbatim* from the Wolf Creek Up-rate License Report [Ref. A1] for several topic areas covered in that license amendment request and the associated review documentation provided in the NRC Safety Evaluation Report [SER, Ref. A2].

A.1 Main Feedwater Performance

Wolf Creek Submittal Information: The following is abstracted from Section 3 (Safety Evaluations), subsection 3.6 (Balance of Plant), part 3.6-IV-4 (Main Feedwater) of the Wolf Creek Up-rate License Report [Ref.-A1]:

The main feedwater system delivers feedwater at the required temperature and pressure from the condensate system to the four steam generators. The feedwater system includes feedwater piping, steam generator feed pumps, isolation and control valves, high pressure feedwater heaters and flow transmitters. Each major system component was evaluated to determine their adequacy for power re-rating to the proposed 3565 MWt reactor power level as follows:

Feedwater Piping: Feedwater is supplied to the four steam generators by four 14" diameter carbon-steel lines. Each line is anchored at the containment wall and designed with sufficient flexibility to provide for relative movement of the steam generators due to thermal expansion. At the most limiting power re-rate condition (Case AB), the feedwater flow will increase by less than 1.5-% above the VWO (valve wide open) flow of 15.85E+6 lb/hr to 16.08E+6 lb/hr. The highest velocity in the feedwater piping at the most limiting power re-rate condition is approximately 28 ft/sec., which is within the recommended design limit of 30 ft/sec. For power re-rate, only the Main Feedwater (MFW) system temperature increases, as compared to the full (100%) power or VWO conditions. The process temperatures for all other plant systems remain the same, or decrease slightly. Since the original MFW system piping stress analyses utilized a feedwater temperature for thermal analyses exceeding the estimated power re-rate value of 446 F, the proposed power re-rate will have no affect on existing MFW System stress analyses.

Steam Generator Feedwater Pumps and Turbine Drivers: Two 2/3-capacity turbine-driven steam generator feedwater pumps (SGFPs), piped in parallel, are provided to supply pre-heated feedwater to the steam generators. At the most limiting power re-rate condition (Case 1AB), the required feedwater pump flow increases to 16.08E+6 lb/hr. This represents an increase of less than 1.5% over the VWO feedwater flow rate of 15.85xE+6 lb/hr. At this condition, the SGFPs require approximately 121 psi of NPSH. Based on a hydraulic review of the main feedwater system, there will be approximately 312 psi of NPSH available to the suction of the SGFPs and, therefore, an adequate margin to support SGFP operation.

The Main Feedwater Pump Turbines (MFPT) were incorporated into the turbine-cycle heat balance calculations. The brake horsepower required to drive the SGFPs is directly proportional to the SGFP flow and the total dynamic head required. Due to the reductions in steam generator pressures at the power re-rate conditions, as compared to the 100% power condition, in combination with the increase in flow rate, the brake horsepower required from the MFPTs to drive the SGFPs will not change significantly. The existing SGFP turbines are adequately sized to drive the main feedwater pumps at the proposed power re-rate conditions.

HP Feedwater Heaters # 5, 6, and 7: Two parallel strings of three HP heaters are provided to heat the feedwater for supply to the steam generators. The HP feedwater heaters were originally sized based on 120-% of VWO (valve wide open) flow plus margins for fouling. HP feedwater heater # 5 can accommodate the additional duty and extraction steam/drain flows associated with one train of LP heaters out of service and 1/3 of the VWO condensate flow through the LP heater bypass valve.

At the most limiting power re-rate condition (Case 1AB), the total feedwater flow requirement of 16.08E+6 lb/hr will increase by less than 1.5% above the VWO flow of 15.85E+6 lb/hr and thus the HP feedwater heater tube side velocities and pressure drop will increase slightly. However, the feedwater heater tube side velocities are still within Heat Exchanger Institute (HEI)) guidelines. The increase in feedwater heater tube side pressure drop has also been accounted for in the evaluation of the main feedwater pumps and found to be acceptable. In addition, a review of the design tube side and shell side flow rates against the most limiting power re-rate flow rates indicates that the power re-rate flows are bounded by the design flow rates. The duty on the HIP feedwater heaters will change due to the heating steam and feedwater flow requirements at power re-rate conditions. The HP feedwater heaters were modeled in the PEPSE program demand mode. Based on the re-rate HP feedwater flow requirements, extraction steam conditions, and HP feedwater heater design data, the terminal temperature difference (TTD), drain cooler approach (DCA) and heat transfer duty of each of the existing HP feedwater heaters were estimated, and are summarized below.

HP Feedwater Heater Design and Calculated Performance Data

Case Conditions	Heater # 5	Heater # 6	Heater # 7
Terminal Temperature Difference (F)/Drain Cooler Approach (F)			
Design	5.00/N/A	5.00/10.0	8.00/10.0
Case 1	4.9/N/A	5.1/10.2	8.2/9.8
Case 2	5.0 /N/A	5.2/10.2	8.2/9.7
Case 3	4.8 /N/A	5.1/10.1	8.2/9.7
Heat Transfer Duty (Btu/hr)			
Design	2.46E+8	3.26E+8	3.35E+8
Case 1	2.29E+8	3.52E+8	3.43E+8
Case 2	2.34E+8	3.54E+8	3.45E+8
Case 3	2.42E+8	3.57E+8	3.46E+8

Terminal Temperature Difference (F)=(steam inlet Sat. Temp. - Feedwater outlet Temp.)
Drain Cooler Approach = (Drain Outlet Temp. - Feedwater Inlet Temp.)

Based on the above TTDs and the final feedwater temperatures, it can be concluded that the HP feedwater heaters are adequate to support power re-rating.

HP Feedwater Heaters #5, #6 and #7 Shell Side Relief Valves: As noted above, the HP design feedwater heater shell side and tube side flow rates envelope the power re-rate flow rates. Therefore, the existing shell side relief valves are adequately sized to protect the HP feedwater heaters from over-pressurization at the proposed power re-rate conditions.

Main Feedwater Isolation Valves (MFIVs): One MFIV is installed in each of the four main feedwater lines to isolate the safety-related portions from non-safety-related portions of the system and to prevent uncontrolled blowdown from more than one steam generator in the event of a feedwater line rupture.

The MFIVs were reviewed based on the flow rates and process pressures at the most limiting power re-rate condition (Case 1AB). The MFIVs were originally designed based on a mass flow rate approximately 3-1/2 times the VWO flow of 15.85E+6 lb/hr in the forward direction. Since the increase in feedwater flow to 16.08E+6 lb/hr is small (<1.5-%) and the required feedwater pressure decreases for all of the power re-rate cases, power re-rate will have no impact on the capability of the existing MFIVs to perform their intended function. Therefore, it is concluded that the MFIVs are adequate to support power re-rate.

Main Feedwater Control Valves (MFCV): The MFCVs, in conjunction with main feedwater pump turbine speed control, provide for adjustment of steam generator water level. For the most limiting power re-rate condition (Case 1AB), the feedwater flow requirement through each MFCV will increase by less than 1.5-%, as compared to the VWO (design) case, and the feedwater pressure required to the steam generators will decrease. The flow increase will require slightly more pressure drop across each MFCV during power re-rate operations. However, the valves have been reviewed, and are adequately sized to support operation at the power re-rate conditions.

Main Feedwater Flow Meters: The feedwater flow transmitters are provided across each feedwater flow element located in each main feedwater supply line to transmit input signals to the feedwater control system for feedwater control valve throttling. Based on a review of the instrument ranges of the flow transmitters, as compared with the most limiting power re-rate conditions (Case 1AB), the existing instruments have sufficient range to function properly for power re-rating.

NRC Evaluation of Adequacy of Licensee Submittal: The following is abstracted from the NRC Safety Evaluation Report for the Wolf Creek Up-rate Application [Ref. A2].

The main feedwater system delivers feedwater, at the required pressure and temperature, to the four steam generators. The safety-related portions of the system ensure isolation capability and provide a path to permit the addition of auxiliary feedwater for reactor cooldown following design basis transients. The licensee's evaluation shows that the existing design basis for the main feedwater isolation valves and main feedwater bypass isolation valves is not significantly affected by operation at the re-rate conditions. The piping configurations associated with the feedwater and auxiliary feedwater systems do not change as a result of the re-rate conditions.

The ability of the auxiliary feedwater system to perform its heat removal function was addressed by the licensee. The staff finds that the safety functions of the feedwater system will continue to be satisfied during operation at the re-rate conditions.

A.2 Main Steam (AB) Performance

Wolf Creek Submittal Information: The following is abstracted from Section 3 (Safety Evaluations), subsection 3.6 (Balance of Plant), part 3.6-IV-1 (Main Feedwater) of the Wolf Creek Up-rate License Report [Ref.-A1]:

The function of the main steam system is to supply steam, generated in the steam generators, to the turbine-generator system and auxiliary systems for power generation. The system provides an assured source of steam to operate the turbine driven auxiliary feedwater pump during emergency conditions. The following main steam system components were reviewed to determine their capabilities to support plant operation at the proposed power re-rate conditions:

Main Steam System Component Design Parameters

Components	Original Design Parameters
Main Steam Safety Valves (MSSVs)	796,500 lb/hr per valve
Main Steam Isolation Valves (MSIVs)	3,964,231 lb/hr @ 1.05 psia
Power Operated Relief Valves (PORVs)	594,642 lb/hr per valve
Turbine Bypass Valves (TBVs)	528,571 lb/hr per valve
Main Steam Flow Transmitters	-52 thru 343 inch
Main Steam Piping	15,136,752 lb/hr, 100-% flow 15,850,801 lb/hr, valve wide open

Main Steam Safety Valves (MSSVs): Each of the four main steam lines is provided with five spring-loaded safety valves. The first safety valve set pressure is 1185 psia (1200 psia), which corresponds to the steam generator design pressure, per ASME Code requirements. The remaining safety valves are set at higher pressures so all safety valves are open at full relief capacity without exceeding 110-% of the steam generator design pressure.

The MSSVs are required to pass 105-% of the Engineered Safeguards Design (END) steam flow at a pressure not to exceed 110-% of the steam generator design pressure. Westinghouse recommends the MSSVs be capable of relieving 105-% of the maximum power re-rate main steam flow, which is 15,940,000 lb/hr for Case 1 AB, or, 1.05 x 15,940,000 lb/hr (16,737,000 #/hr). The combined relieving capacity of the existing safety valves is equivalent to 18,229,608 lb/hr, and thus, sufficient for power re-rate. Based on the preceding, the existing Main Steam Safety Valves are adequate to support power re-rate.

Main Steam Isolation Valves (MSIVs): To isolate the non-safety related portions of the main steam system and to assure that steam is available to operate the turbine-driven auxiliary feedwater pump for reactor cooldown following a loss of main feedwater, a main steam isolation valve is provided in each of the four main steam lines outside of the containment, downstream of the MSSVs. The valves were originally designed to pass 105-% (VWO) steam flow and close against the steam generator no-load pressure of 1107 psia.

The MSIVS flow capacity and capability to close were reviewed at the most limiting power up-rate condition, Case 1AB. Due to the increase in steam flow rates and the decrease in main steam pressure, the steam velocity through the MSIVs will increase at power re-rate conditions. However, the increase in steam velocity will not adversely affect the MSIVs' operation. The MSIVs were originally designed based on an inlet steam pressure equal to the steam generator no-load pressure of 1107 psia, and a mass flow rate approximately four times the VWO flow in the forward direction, and approximately ten times the VWO flow in the reverse direction. Since the increase in steam flog due to power re-rating is small (~0.57% (15.85E+10 lb/hr @ VWO vs. 15.94E+10 lb/hr @ Case 1AB) and the steam line pressure decreases, power re-rate will not adversely impact MSIVS closure times, or the capability of the valves to close and remain closed. Therefore, it can be concluded that the MSIVs are adequate to support power re-rate.

Power Operated Relief Valves (PORVs): One power operated relief valve is provided on each of the four main steam lines to control steam generator pressure during startup, load changes and shutdown, when the main steam isolation valves are closed or when the turbine bypass system is not available. The valves were originally sized to relieve 15-% of VWO steam flow (15.85E+10 lb/hr) at steam generator no-load pressure (1107 psia), and to pass sufficient flow at all steam generator pressures to achieve a 50 F per hour plant cooldown.

Westinghouse recommends that the minimum combined relieving capacity of the PORVs be 10-% of the plant design steam flow rate. This represents a maximum steam flow capacity of 398,500 lb/hr per steam line, at the maximum power re-rate condition (Case 1AB). Since the existing PORVs are sized to relieve 15-% of the VWO main steam flow, or 594,642 lb/hr per steam line, the existing PORVs are adequately sized for the proposed power re-rate conditions.

The set pressure for the PORVs shall be between the no-load steam generator pressure, which is 1107 psia for the power re-rate, and the set pressure of the lowest set main steam safety valve which is 1185 psia. The existing PORV set pressure of 1125 psia falls within these values, and therefore, the existing set pressure is adequate for power re-rate.

Turbine Bypass Valves (TBV): The turbine bypass valves are provided to enable the NSSS to follow turbine load reductions by dumping steam to the HP, IP, and LP condensers. The twelve turbine bypass valves were originally sized to pass a total of 40% of VWO steam flow, and thus, permit the turbine to take a 50- load rejection without a reactor trip. These valves also allow a turbine and reactor trip from full load without lifting the main steam safety valves.

Each turbine bypass valve was originally designed with a flow capacity of 528,571 lb/hr at 970 psia, and a maximum capacity of 970,000 lb/hr at 1200 psia. In order to satisfy their functional design basis, the turbine bypass valves must be capable of relieving 40-% of the power re-rate full load main steam flow, at a pressure below the set point of the lowest set MSSVS.

Based on a review of the turbine bypass valve relieving capacity at each of the power re-rate conditions, the valves are adequately sized to satisfy this design basis. Therefore, the turbine bypass valves are adequate to support power re-rate. Main Steam Flow Transmitters (FT-512 through FT-543)

Two main steam flow transmitters are provided on each main steam line to transmit input signals to the feedwater control system to provide for feedwater pump speed control and feedwater control valve throttling. Based on a comparison of the instrument ranges to the most limiting power re-rate process conditions (Case I NB), the existing instruments have sufficient range to operate properly at the proposed power re-rate conditions.

NRC Evaluation of Adequacy of Licensee Submittal: The following is abstracted from the NRC Safety Evaluation Report for the Wolf Creek Up-rate Application [Ref. A2].

The main steam system dissipates energy generated by the reactor core to the turbine generator and auxiliary steam loads, the main condenser via the steam dump valves, or to the atmosphere via atmospheric relief valves or main steam safety valves. Isolation of the main steam system is achieved by the main steam isolation valves and main steam bypass isolation valves. The licensee evaluated the capability of the main steam system components to perform their design functions under the proposed re-rate conditions. The licensee determined that the existing set-points and capacity of the main steam safety valves are adequate to prevent exceeding 110 percent of design pressure of the main steam system under the most limiting transient. The set-point and capacity of the atmospheric relief valves were found to remain adequate to control the design load shed of 10-% rated thermal power. In addition, the atmospheric relief valves were found to have adequate capacity to achieve a 50 F/hr cooldown if the main condenser was unavailable. The main steam isolation valves were evaluated to ensure the valves will continue to perform their isolation function under the maximum differential pressure conditions and within the time limits assumed in the safety analysis. The staff concludes that the existing main steam system components are adequate to perform their safety functions under the re-rated plant conditions.

A.3 Control Rod Drive Mechanism

Wolf Creek Submittal Information: The following is abstracted from Section 3 (Safety Evaluations), subsection 3.5 (Reactor Coolant System Components and Fluid Systems Evaluation), part 3.5.1.4 (Control Rod Drive Mechanism) of the Wolf Creek Up-rate License Report [Ref.-A1]:

The original design report establishes the structural integrity of the pressure boundary components of the Model L-106A1 Control Rod Drive Mechanism (CRDM) including the seismic sleeve and the capped latch housing assembly for the Wolf Creek Generating Station as required by Section-III of the ASME Boiler & Pressure Vessel Code, 1974 Edition through Winter 1974 Addenda. To determine the effect of the re-rating the revised primary side parameters and the associated nuclear steam supply system (NSSS) design transients were compared to those analyzed in the original design report. The evaluation results indicate that the re-rating temperature ranges have a diminishing and insignificant effect on the thermal analysis and that the revised design transients are bounded by those analyzed in the original design report.

Therefore, the Wolf Creek Model L-106A1 control rod drive mechanism remains in compliance with the Westinghouse and industry codes and standards, which were originally applicable when Wolf Creek was initially licensed.

NRC Evaluation of Adequacy of Licensee Submittal: The following is abstracted from the NRC Safety Evaluation Report for the Wolf Creek Up-rate Application [Ref. A2]. Control Rods

The licensee evaluated the adequacy of the Control Rod Drive Mechanisms (CRDMs) by comparing the design bases input parameters with the operating conditions for the proposed re-rate. The licensee stated that the re-rate conditions would have an insignificant impact on the original design bases analyses for the CRI)Ms. 'The staff has reviewed the licensee's evaluation and concurs with the licensee's conclusion that the current design of the control rod drive mechanisms would not be impacted by the re-rate.

A.4 References

A1. Wolf Creek Nuclear Operating Corp, *"Power* Up-rate Report for the Wolf Creek Generating Station", NRC Docket #: 50-482, (Dec. 1992).

A2. U. S. Nuclear Regulatory Commission, "*Safety Evaluation Report for the Wolf Creek Generating Station*", (Nov. 10, 1993).

Synergistic Safety Issues Related to Reactor Power Uprates

by

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During the past several decades the NRC has reviewed and approved in excess of 30 licensee requests for power uprates. Each request has been evaluated to assure that current regulatory requirements are satisfied; nevertheless, certain synergistic processes exists that need to be adequately covered in such reviews. Specifically higher power levels when combined with system/component degradation via plant aging and fuel life extensions to high burnup may adversely impact plant safety margins. Evidence of these effects stem from recent events noted for operating plants, including failure to fully insert control rods in high-power/high-burnup fuel assemblies and piping failures. This paper examines the potential for synergistic effects [*synergistic*---the cooperative action of discrete agencies such that the total effect is greater than the sum of the individual effects] and the need to consider such in the uprate review process.

A utility seeking a power uprate will submit a request in the form of a Licensing Amendment Report (LAR), which contains information similar to that found in the original FSAR but at the uprated power level. The LAR centers on a re-evaluation of design basis accidents (DBA) and off-normal transients, and the adequacy of safety related systems to perform their intended function at the higher power. Information presented in the LAR is reviewed by the NRC staff and its findings are reported in an uprate Safety Evaluation Report (SER). The NRC review encompasses consideration of any new or unreviewed safety concerns in accordance with 10CFR-Part 50.59, as well as review of changes to plant technical specification. The uprate application is approved if the case has been made that applicable regulations are satisfied at the uprated power.

The uprate applications reviewed in this study include that for the Brunswick, Maine Yankee, North Anna, Surry and Wolf Creek plants. A review of the LARs and SERs for these plants revealed little documentation with regards to consideration of potential synergistic effects of high core power densities when combined with component aging and/or high burnup fuel effects. A review of operational events for uprated plants however shows evidence that such synergistic effects may occur. Examples include the control rod insertion problems noted at the Wolf Creek and North Anna plants, both having received power uprate approvals in the range of 4-5%. At Wolf Creek five control rods failed to fully insert during scram from full power. The affected control rods involved Westinghouse Vantage-5H fuel assemblies with burnups greater than 47,6000 MWD/t-U. Root cause analysis indicate Zircaloy guide tube distortion due to irradiation induced growth. Since irradiation growth in metals is influenced by neutron energy spectrum, power level, and total exposure (burnup) effects, synergistic processes are evident. Control rod sticking problems have also been noted at North Anna-1. An examination the Wolf Creek and North Anna uprate documentation (LARs and SERs) for control rod behavior did not reveal consideration of the effects of higher power level when combined with elevated fuel burnup conditions. Other incidents include power offset anomalies for long-cycle/high-power cores tied to crud buildup on high-burnup fuel rods. The crud appears to getter boron causing a distortion of the axial power profile, particularly in high power assemblies, again indicative of potential synergistic effects.

Aged reactor components and systems, in combination with high core power densities, may likewise produce degradation that is greater than the sum of the individual effects. Research has shown that pipe corrosion is often exacerbated at increased fluid velocities, indicative of a synergistic corrosion/erosion process. The main feedwater pipe break at Surry-2 in 1986 provides an example for such a synergistic process. During at-power operation a main feedwater pipe ruptured at the Surry-2 plant, resulting in four site-worker fatalities due to release of scalding steam from the ruptured pipe. Post-accident investigations revealed feedwater pipe thinning and catastrophic rupture due to combined corrosion/erosion effects. Although the Surry-2 plant was operating at its original power rating, a linkage to uprated power conditions is made since they often involve an increase in feedwater flow. Other aging factors include vibrational fatigue, particularly for pumps and valves in the vicinity of moving machinery. Higher coolant velocities associated with power uprates would tend to aggravate vibrational fatigue.

Inadequacies were also noted in regards to the Maine Yankee and Brunswick uprate applications, where deficient licensee submittal information was not uncovered during the initial review process by NRC. Both incidents indicate a need for independent agency thermal-hydraulic and neutronic analysis capabilities, to verify the accuracy of licensee submittal information and analysis. NRC in-house computational efforts would go a long way in providing an independent check and verification of what is now essentially a licensee effort. In view of such observations the following recommendations are made:

- NRC should issue a Standard Review Plan (SRP) for power uprate applications, which should include acceptance criteria which consider the influence of synergistic effects, specifically high fuel burnup levels and component/system aging effects in combination with uprated power conditions. The NRC is in the process of developing a power uprate SRP.

- NRC uprate review procedures should include requirements for independent NRC staff analysis (i.e. thermal-hydraulic and neutronic code predictions) and verification of uprate plant predictions submitted by the licensee. The results of these NRC audit calculations should be included in the SER for each uprate application and include comparisons with licensee submittal analysis.

- A comparison of probabilistic safety measures (e.g. CDF, QHO, LERF) at the uprated and prior power levels is recommended for uprate applications.