DRAFT-Rev-0

Perspectives and Insights from ACRS Review of Staff's Safety Evaluations of Power Uprate License Amendment Requests

Prepared By
Hossein P. Nourbakhsh
Senior Technical Advisor
Office of Advisory Committee on Reactor Safeguards

January 2010

Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

ABSTRACT

This report has been prepared for use by the NRC Advisory Committee on Reactor Safeguards (ACRS) in its ongoing review of staff's safety evaluations of power uprate applications. The regulations and guidance documents related to power uprate and the role of ACRS in approving a change to a plant's power level have been discussed. The Committee's observations and recommendations on staff's safety evaluations of power uprate applications have been summarized to provide insights and perspectives on previous Committee's review of power uprate applications. An overview of international perspectives on power uprate has also been presented.

The views expressed in this paper are solely those of the author and do not necessarily represent the views of the ACRS.

TABLE OF CONTENTS

FIGURES

TABLES

ABBREVIATIONS

ACRS	Advisory Committee on Reactor Safeguards
AEC	Atomic Energy Commission
ASLBP	Atomic Safety and Licensing Board Panel
ATWS	Anticipated Transient without Scram
BWR	Boiling Water Reactor
CFR	Code of Federal regulations
DBA	Design Basis Accident
EDO	Executive Director for Operations
EPU	Extended Power Uprate
ESF	Engineered Safety Feature
FSAR	Final Safety Analysis Report
GENE	General Electric Nuclear Energy
IAEA	International Atomic Energy Agency
LOCA	Loss-of-Coolant Accident
LOSP	Loss of Offsite Power
LWR	Light Water Reactor
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
PRA	Probabilistic Risk assessment
PWR	Pressurized Water Reactor
RCS	Reactor Coolant System
RIS	Regulatory Issue Summary
RPV	Reactor pressure Vessel
RS	Review Standard
SBO	Station Blackout
SER	Safety Evaluation Report
SGTR	Steam Generator Tube Rupture
SPU	Stretch Power Uprate
SRM	Staff Requirements Memorandum
SRP	Standard Review Plan

1 INTRODUCTION

In the U.S., NRC regulates the maximum power level at which a commercial nuclear power plant may operate. This power level is used, with other data, in many of the licensing analyses that demonstrate the safety of the plant and is included in the license and technical specifications for the plant. The process of increasing the maximum power level at which a commercial nuclear power plant may operate is called a power uprate. Power uprates are submitted to NRC as license amendment requests. Requesting and approving a plant's power uprate is governed by Sections 90, 91, and 92 , Part 50 of the Title 10 of *Code of Federal Regulations* (10 CFR 50.90, 50.91, and 50.92) [1].

Utilities have been using power uprates since the 1970s as a way to increase the power output of their nuclear plants. To increase the power output of a reactor, typically more highly enriched uranium fuel and/or more fresh fuel is used. This enables the reactor to produce more thermal energy and therefore more steam, driving a turbine generator to produce electricity. In order to accomplish this, components such as pipes, valves, pumps, heat exchangers, electrical transformers and generators, must be able to accommodate the conditions that would exist at the higher power level. In some instances, licensees will modify and/or replace components in order to accommodate a higher power level. Depending on the desired increase in power level and original equipment design, this can involve major modifications to the plant such as the replacement of main turbines. All of these factors must be analyzed by the licensee as part of a request for a power uprate, which is accomplished by amending the plant's operating license. The analyses must demonstrate that the proposed new configuration remains safe and that measures continue to be in place to protect the health and safety of the public. These analyses, which span many technical disciplines and may be complex, are reviewed by the NRC before a request for a power uprate is approved.

Power uprates can be classified in three categories: (1) measurement uncertainty recapture power uprates, (2) stretch power uprates, and (3) extended power uprates. Measurement uncertainty recapture power uprates are on the order of 1.5 percent and are achieved by implementing enhanced techniques for calculating reactor power. This involves the use of state-of-the-art feedwater flow measurement devices that reduce the degree of uncertainty associated with feedwater flow measurement and in turn provide for a more accurate calculation of power. Stretch power uprates are typically on the order of 7 percent and usually involve changes to instrumentation setpoints. Stretch power uprates do not generally involve major plant modifications. This is especially true for boiling-water reactor (BWR) plants. In some limited cases where plant equipment was operated near capacity prior to the power uprate, more substantial changes may be required. Extended power uprates are usually greater than stretch power uprates and are expected to be submitted for increases as high as 20 percent. Extended power uprates usually require significant modifications to major balance-of-plant equipment such as the high pressure turbines, condensate pumps and motors, main generators, and/or transformers.

ACRS reviews the power uprates that are amounting to power increase greater than 5 percent above originally licensed value. Since 1998, ACRS has reviewed fifteen applications for power uprates. ACRS was instrumental in the staff development of a

review standard for extended power uprates [2]. This standard, issued in December 2003, is a first-of-a-kind document that provides a comprehensive process and technical guidance for reviews by the NRC staff, and also provides useful information to licensees considering applying for an extended power uprate.

This report has been prepared for use by the Committee in its ongoing review of staff's safety evaluations of power uprate applications. A number of reference materials, including regulations related to amendment of a license (10 CFR 50.90, 50.91, and 50.92), review standard for extended power uprates (RS-001) [2], power uprates section of the NRC Public Website [3], and the ACRS reports on the past reviews of power uprate applications were reviewed for the preparation of this report.

The report begins with an overview of regulations and guidance documents related to power uprate as well as the role of ACRS in power uprate regulatory process. It then summarizes the Committee's observations and recommendations on the previous staff's safety evaluations of power uprate license amendment requests. The report also presents an overview of international perspectives on power uprate.

2 REGULATIONS RELATED TO THE POWER UPRATE REQUESTS AND REVIEW PROCESSES

The process for amending commercial nuclear power plant licenses and technical specifications related to power uprates is the same as the process used for other amendments. Therefore, power uprate requests are submitted to NRC as license amendment requests. This process is governed by 10 CFR 50.90, 50.91 and 50.92.

According to 10 CFR 50.90, whenever a holder of a license, including a construction permit and operating license under this part, and an early site permit, combined license, and manufacturing license under part 52 of this chapter, desires to amend the license or permit, application for an amendment must be filed with the NRC, as specified in 10 CFR 50.4 or 10 CFR 52.3, as applicable, fully describing the changes desired, and following as far as applicable, the form prescribed for original applications. 10 CFR 50.91 establishes the procedural requirements for amendment to an operating license or a combined license.

According to 10 CFR 50.92, in determining whether an amendment to a license, will be issued to the applicant, the NRC will be guided by the considerations which govern the issuance of initial licenses to the extent applicable and appropriate. 10 CFR 50.92 states that the NRC will be particularly sensitive to a license amendment request that involves irreversible consequences (such as one that permits a significant increase in the amount of effluents or radiation emitted by a nuclear power plant). 10 CFR 50.92 also states that the NRC may make a final determination, under the procedures in § 50.91, that a proposed amendment to an operating license involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

> (1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or
>
> (2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or
>
> (3) Involve a significant reduction in a margin of safety.

The NRC issued Review Standard (RS)-001, Revision 0, "Review Standard for Extended Power Uprates," in December 2003 [2]. ACRS was instrumental in the staff development of this review standard. This review standard establishes standardized review guidance and acceptance criteria for the staff's reviews of EPU applications to enhance the consistency, quality, and completeness of reviews. It serves as a tool for the staff's use when processing EPU applications in that it provides detailed references to various NRC documents containing information related to the specific areas of review. The development of this review standard included an evaluation of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (SRP) [4],

In addition to ensuring that a license amendment request complies with the NRC regulations and other requirements, it is also the staff's responsibility to consider the

risk aspects of a license amendment request [5-6]. The use of risk information is clear when the licensee or the NRC designates the submittal as a “risk-informed” license application. Guidance is also provided to the staff in Appendix D of Chapter 19 of the Standard Review Plan (SRP) [4] as to the “special circumstances” under which a detailed risk review may be necessary, even for license applications that are not designated as being risk-informed. This process is also described in Regulatory Issue Summary (RIS) 2001-002, “Guidance on Risk-Informed Decisionmaking in License Amendment Reviews,”[6]. Special circumstances are defined in the above guidance as “conditions or situations that would raise questions about whether there is adequate protection, and that could rebut the normal presumption of adequate protection from compliance with existing requirements. In such situations, undue risk may exist even when all regulatory requirements are satisfied.”

In addition to its mission of protecting public health and safety under the Atomic Energy Act, the NRC is charged with protection of the environment in the use of nuclear materials. Pursuant to 10 CFR 51.21, 51.32, 51.33, and 51.35, a draft Environmental Assessment and finding of no significant impact should be prepared and be published in the Federal Register. The draft Environmental Assessment provides a 30-day opportunity for public comment. Based on the final Environmental Assessment, the Commission should determine that the issuance of this amendment will not have a significant effect on the quality of the human environment.

3 THE ROLE OF ACRS IN POWER UPRATE REVIEW PROCESS

According to 10 CFR 50.58 an application for an amendment to a construction permit or operating license for a facility which is of a type described in § 50.21(b) or § 50.22, or for a testing facility shall be referred to the ACRS for a review and report. Figure 1 illustrates the review process and the interrelationships among various review activities.

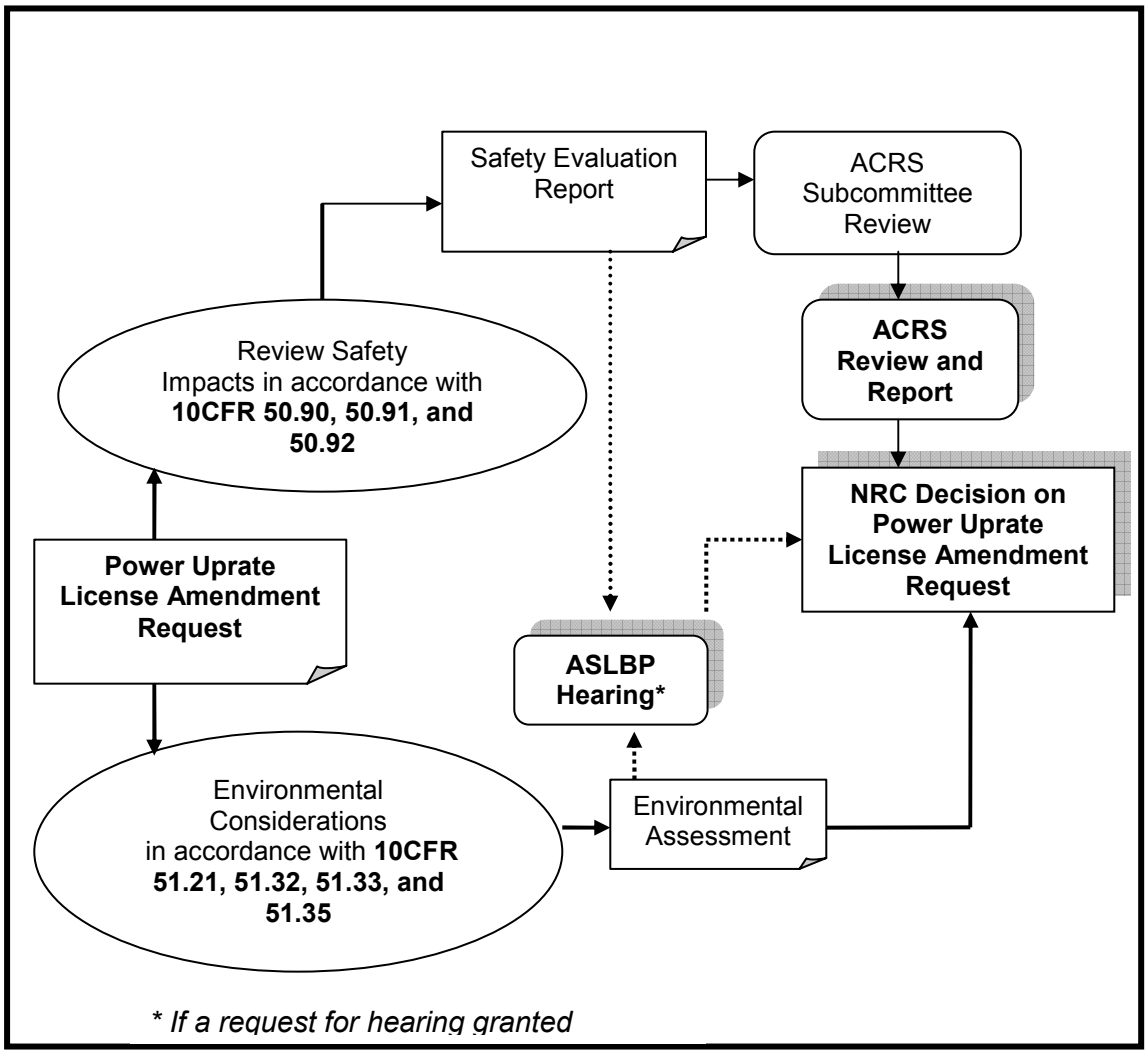

Figure 1: The Power Uprate Review Process

The NRC Office of Nuclear Reactor Regulation (NRR) reviews the power uprate license amendment request and supporting documentation. The review results in a safety evaluation report. Pursuant to 10 CFR 51.21, 51.32, 51.33, and 51.35, NRC

also prepares an environmental assessment to determine that the issuance of amendment will not have a significant effect on the quality of the human environment.

ACRS reviews the staff's safety evaluations of Extended Power Uprate (EPU) and Stretch Power Uprate (SPU),(amounting to power increase greater than 5 percent above originally licensed value) applications. An ACRS review is essential, given the potential safety implications of increasing the maximum power level at which a commercial nuclear power plant may operate. ACRS review begins shortly after safety evaluation report is prepared by the staff. In-depth reviews are done by the ACRS Power Uprate Subcommittee. With input from Subcommittee members, Subcommittee Chairman develops proposed ACRS position. Briefings by the licensee and the NRC staff are provided to both the Subcommittee and Full Committee. ACRS positions are developed after extensive deliberations by the full Committee. When the Committee has completed its review, its report is submitted to the Commission. At times, ACRS issues "interim" letters to identify issues of concern and items for which additional information, discussions, and clarifications are needed.

When the request for power uprate license amendment is submitted, there is an opportunity for individuals or groups to petition for a hearing to address specific issues related to either plant safety or environmental impacts. If granted, a hearing is held and the decision of the Atomic Safety and Licensing Board Panel (ASLBP) is presented to the Commission for its consideration in making a decision on issuing the amendment.

4 INSIGHTS FROM PREVIOUS ACRS REVIEW OF STAFF'S SAFETY EVALUATIONS OF POWER UPRATE REQUESTS

The ACRS has contributed significantly to the success of the power uprate program by establishing expectations on the quality of the power uprate license amendment requests and supporting documentations.

In 1991, General Electric Nuclear Energy (GENE) initiated a power uprate program to support BWR plant licensees for increasing rated core power by up to 5 percent. In 1992, the Committee reviewed the initial GENE power uprate program and the application by the Detroit Edison Company for a power level increase for the Fermi nuclear power plant, Unit 2. In its September 17, 1992 report, the Committee endorsed the GENE generic program associated with the 5 percent power level uprates and concluded that a 5 percent uprate did not pose a significant increase in risk. It was recognized that any power uprate would in some way erode safety margins and that, although 5 percent uprates were acceptable for all BWRs, the ACRS recommended that any uprates beyond that be given additional review and justification.

In 1995, GENE initiated the "extended" power uprate program. The word "extended" was used to distinguish this program from the initial power uprate program. The extended uprate program addressed additional power uprates greater than 5 percent and up to 20 percent of rated core power. Licensees were to make individual decisions on the magnitude of power uprates. The Monticello Nuclear Generating Plant was the lead plant for the extended power uprate program. The Northern States Power Company submitted an application for a power level increase of 6.3 percent for the Monticello Plant.

During the early ACRS reviews of EPUs, the Committee indicated that the staff's documentation of its reviews in the safety evaluations should provide more details concerning the scope and focus of the reviews as well as the criteria used to reach conclusions. The ACRS further indicated that development of an SRP would help ensure adequate reviews by the staff of future power uprate applications and would also clarify to the public and licensees the acceptance criteria for power uprate applications.

During the December 5, 2001, Commission meeting, the ACRS recommended that the staff develop an SRP for power uprates. As a result, in an SRM dated December 20, 2001 [7], the Commission directed the staff to review the ACRS recommendation to develop a standard review plan to improve the effectiveness of power uprate reviews. The staff evaluated the ACRS recommendation to develop an SRP for improving the effectiveness of power uprate reviews and concluded that a greater level of standardization will improve the effectiveness of EPU reviews [8].

In its September 24, 2003 report to the Commission on draft final Review Standard for Extended Power Uprates, RS-001, the ACRS commended the staff for the development of an excellent review standard. The Committee also expressed a concern about synergistic or compounding effects of uprates with other regulatory actions. The Committee stated that while such effects are difficult to identify explicitly, the application of the Review Standard will help call attention to such effects. This is

particularly true for areas with materials concerns where flow accelerated corrosion, fluid structure interaction, fatigue, and stress corrosion cracking can interact and shorten component life.

To date, the NRC has approved power uprate license amendment applications to increase nuclear power generation for over 5,500 MWe. The power uprate applications reviewed by the Committee are listed in Table 1. The ACRS recommendations, observations, and comments provided during its past reviews of power uprate applications that may have generic implications are presented below.

MONTICELLO NUCLEAR GENERATING PLANT

We particularly endorse the staffs requirement that "each applicant report the effects of the proposed uprate on its core damage frequency and frequency of large magnitude radioactive release." We believe that the appropriate process for making decisions related to power uprate applications is that outlined in Regulatory Guide (RG) 1.174 related to requests for changes to the licensing basis. With the addition of an analysis for core damage frequency (CDF); large, early release frequency (LERF); and the changes associated with the uprate (ΔCDF and ΔLERF), the power uprate program will provide the information required to utilize the RG 1.174 process, including that associated with all the deterministic analyses made as part of a safety evaluation report. **(July 24, 1998)**

The staffs recommendation for approval of the power level increase for the Monticello plant is based partly on the IPE that "meets the requirements of GL [Generic Letter] 88.20." It is not clear to us that this standard for IPEs is also the appropriate standard for a PRA on which to base power uprate decisions. A justifiable decision is needed from the staff on the quality standard required for PRAs to assist decisionmaking on power uprate requests. Additional guidance for the applicant is also needed. **(July 24, 1998)**

In any future power uprate application, the staff should require that bounding estimates be made for the contributions from any missing elements of the PRA, especially for the contributions from shutdown, low power, and external events. **(July 24, 1998)**

We are concerned about the concept that seemed to be implied in the application and the staff's review documents that, because better calculations are now possible, greater margins exist. The margin is inherent in the design and is what it is, regardless of the calculational ability. These margins compensate for aleatory and epistemic uncertainties in the determination of the risk status. We believe that any power uprate has the effect of eroding the margins. This is the reason for our recommendation that the NRC staff guide its decisions on power uprates by the intent of the RG 1.174 process, which provides the appropriate rationale for justifying decreases in margins. **(July 24, 1998)**

Table 1 Power Uprate License Amendment Requests Reviewed by ACRS

Plant	Design	Power MWt	% Uprate	Subcommittee Meeting Date	Full Committee Meeting Date	ACRS Report Date	Remarks
Monticello	BWR Mark I	1670	6.3	June 2, 1998	June 3, 1998 July 8, 1998	July 24, 1998	
Hatch 1&2	BWR Mark 1	2463	8	August 27, 1998	Sep. 2, 1998	Sep. 15, 1998	
Duane Arnold	BWR Mark1		15.3	Sep. 26-27, 2001	Oct. 2, 2001	Oct. 17, 2001	Used GE generic methodology (ELTR-1 and ELTR-2)
Dresden 2&3 **Quad Cities 1&2**	BWR/3 Mark I		17 17.8	Oct. 25-26, 2001	Nov. 7, 2001 Dec. 5, 2001	Nov. 13, 2001 Dec. 12, 2001	ELTR-1 and ELTR-2
ANO Unit 2	PWR (CE)		7.5	Feb. 13, 2002	March 7, 2002	March 14, 2002	
Clinton Unit 1	BWR/6 Mark III	2894	20	Feb. 13-14, 2002	March 7, 2002	March 14, 2002	Constant pressure power uprate (CPPU)
Brunswick Units 1&2	BWR/4 Mark I		15	April 23, 2002	May 2, 2002	May 10, 2002	ELTR-1 & ELTR-2 CPPU GE Topical Report (NEDC-33004P)
Waterford Unit 3		3441	8	Jan. 26, 2005	Feb. 10, 2005	Feb. 24, 2005	First use of RS-001

Table 1 Power Uprate License Amendment Requests Reviewed by ACRS (continued)

Plant	Design	Power MWt	% Uprate	Subcommittee Meeting Date	Full Committee Meeting Date	ACRS Report Date	Remarks
Vermont Yankee	BWR/4 Mark I	1592	20	Nov. 29-30, 2005	Dec. 7, 2005	Jan. 4, 2006	
Beaver Valley Units 1&2		2689	8	April 24-25, 2006	May 4, 2006	May 22, 2006	
R.E. Ginna	PWR W-2L	1520	17	March 14-15, 2006 April 2006	May 4, 2006	May 22, 2006	Replaced SGs in 1996 Replaced RVH in 2003 Replaced HP turbine & turbine control valves
Browns Ferry Unit 1	BWR/4 Mark I	3293	5	Jan. 16-17, 2007	Feb. 1, 2007	Feb. 16, 2007	Two step approach
Susquehanna Units 1&2	BWR/4 Mark II	3293	20	Oct 9-10, 2007 Nov. 14, 2007	Dec. 6, 2007	Dec. 20, 2007 June 3, 2008	
Hope Creek	BWR/4 Mark I	3339	15	March 20-21, 2008	April 10, 2008	May 2, 2008	
Millstone Unit 3	PWR	3411	7	July 8, 2008	July 9, 2008	July 23, 2008	Stretch Power Uprate

EDWIN I. HATCH NUCLEAR POWER PLANT UNITS I AND 2

The Southern Nuclear Operating Company, Inc. (*SNC) presented substantial probabilistic risk analyses and uncertainty evaluations to support a risk-informed decision process. These analyses included reviews of initiating event frequencies, equipment failure rates, operator errors, and success criteria. Qualitative arguments were used to establish that the contributions of fire, external events, and events under shutdown conditions to CDF are likely to be acceptably small. The effects of power uprate on the expected value of the large, early-release frequency were shown to be small.* **(September 15, 1998)**

DUANE ARNOLD ENERGY CENTER (DAEC)

Many technical issues must be addressed in an application for power uprate. Of these, we consider five to be especially significant:

1. *Susceptibility of the plant to ATWS (Anticipated Transients Without Scram)*
2. *ATWS recovery*
3. *Reduction in some of the times available for operator actions because of higher decay heat*
4. *Material degradation due to irradiation-assisted stress corrosion cracking (IASCC) of reactor internals and flow-assisted corrosion and fatigue of feedwater piping.*
5. *Containment response to accident events involving higher decay heat levels.* **(October 17, 2001)**

We found it far more difficult to assure ourselves that the DAEC core is susceptible only to global power oscillations and does not need to consider local power oscillations. It was similarly difficult to assure that ATWS recovery methods were applicable to cores with flattened power profiles, that critical human actions had been identified with adequate independence by the staff, and that material degradation sensitivities had been adequately assessed. **(October 17, 2001)**

Many of the challenges that we encountered in our review of the DAEC power uprate application could have been eased if the staff had improved guidance on the detail to be provided in SERs and developed criteria for when independent assessments should complement reviews of applicant submittals. **(October 17, 2001)**

DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3 AND QUAD CITIES NUCLEAR POWER STATION, UNITS 1 AND 2

The GE Topical Report, ELTR-1[9], that supports the EPUs includes the requirement that certain large transient tests be performed to confirm the effectiveness of the implemented plant modifications. GE has since reached the conclusion that these tests are not necessary for power uprates in which reactor steam dome pressure is not changed. The staff agrees with this conclusion for the Dresden and Quad Cities applications. Technical arguments to support this decision are documented in the Safety Evaluation (SE). We concur with the staff's conclusion for these plants. **(December 12, 2001)**

ARKANSAS NUCLEAR ONE, UNIT 2

The process used by the Applicant to perform the Reload Safety Analysis appears to be appropriate. Because this is the first large power uprate for a PWR, the staff should review the Reload Safety Analysis for the transitional core reloads to ensure that the plant will operate in compliance with the regulations. **(March 14, 2002)**

It is difficult to perform a major power uprate in a PWR unless significant modifications are made to the plant. In a PWR, the power is limited by the amount of heat exchange surface. ANO-2 installed larger replacement steam generators that can accommodate the higher thermal power, but, these larger steam generators impose greater accident loads on the containment. The increased energy release during a potential steamline break accident required an increase in the containment building design pressure rating from 54 psig to 59 psig. Instead of modifying the containment building, the Applicant reanalyzed the strength of the containment - considering additional tendons that had not been credited in the original analysis. The containment pressure capability was demonstrated by conducting a pressure test at 68 psig. We conclude that the Applicant's analyses of containment loads and demonstration of the design capability of the containment structure are adequate. **(March 14, 2002)**

CLINTON POWER STATION, UNIT 1

The constant-pressure power uprate produces higher steam and feedwater flows in the plant. The higher flows in the steamlines carrying scavenging steam to the high-pressure feedwater heaters are predicted to increase the flow-assisted corrosion in these lines to as much as 0.070 inches per year. The licensee is persuaded that the predictions of the flow-assisted corrosion rates in these lines with 0.500-inch thick walls are conservative, but acknowledges that the corrosion in these lines will be accelerated by the power uprate. **(March 14, 2002)**

There has been an unfortunate history within the U.S. nuclear industry of pipe ruptures in nonsafety systems because of flow-assisted corrosion. These ruptures have had safety consequences even when they have occurred in lines that are usually found not to have great risk significance. It is important, then, that the licensee's program for monitoring flow-assisted corrosion in steam and feedwater lines be rigorously conducted. It is also important that the staff reviewing the power uprate application have a good process that communicates the importance of the monitoring program to the staff who inspect the uprated plant. **(March 14, 2002)**

The licensee proposes not to conduct the large transient tests called for in the current version of the General Electric extended power uprate methodology. The staff has accepted this proposal and feels confident that analysis methods are adequate to predict plant performance. We have not found a value for these tests that are commensurate with costs and risks and, therefore, support the position not to conduct the large-transient tests. The modifications to the plant proposed by the licensee do not involve changes to the "recirculation runback system." **(March 14, 2002)**

BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2

Carolina Power and Light Company (CP&L) has committed to modify the standby liquid control system (SLCS) in which the boron solution is sufficiently enriched with Boron-10. This modification will allow the shutdown capability to be met in the event of an anticipated transient without scram (ATWS) with the use of only one of the two available SLCS pumps. The licensee calculates that this modification will reduce the plant's internal events core damage frequency (CDF) and large early release frequency (LERF) by 9% and 28%, respectively. Without the use of enriched boron, the ATWS risk increases slightly due to shortened times for operator decisions. Because of the significant safety benefit that is obtained by offsetting the most significant risk increase associated with EPU, we agree that this modification to the SLCS should be implemented. **(May 10, 2002)**

We encourage the staff to continue to pay close attention to the details of core reload analyses at Brunswick and other BWR EPU plants. This is particularly important with regard to the ways that core thermal success criteria will continue to be met as more sophisticated fuel design and reload management techniques are implemented. The staff should assess the need for more detailed thermal-hydraulic models of the core, replacing the current "averaging" approaches, to complement present neutronic analyses that model the wide variations in fuel composition and power level throughout the core. **(May 10, 2002)**

This review demonstrates an inherent problem in the "two-tier" regulatory system. The application for the EPU was not risk-informed, yet a PRA was submitted. This creates a situation in which the PRA is not seriously reviewed, although it is part of the record. Also, the uncertainties in human reliability analysis are significant, but there is no mention of them. The applicant used human reliability models that have not been reviewed by the staff. The staff acknowledges that large uncertainties are present and that the models have not been reviewed. However, the staff concludes that insights regarding the relative importance of operator actions can be gained. In addition, the potential increases in the change in core damage frequency (CDF), that could arise if the PRA were capable of modeling the effect of margin reductions on risk, are not included. **(May 10, 2002)**

PRA quality is essential for risk-informing the regulations. Improvements in PRA quality, such as inclusion of the effects of margin reductions on risk and improving human reliability models, may be discouraged as long as important decisions such as granting power uprates are made by "accepting" PRAs without criticism because the application is not risk-informed. **(May 10, 2002)**

WATERFORD STEAM ELECTRIC STATION, UNIT 3

The application by Entergy for an 8% extended power uprate (EPU) at Waterford 3 should be approved, subject to (1) the staff's approval of the alternate source term (AST) application and (2) documentation of the resolution of the boron precipitation issue during long-term cooling for Waterford 3 by the submittal of the analysis details and their acceptance in the staff's safety evaluation (SE). **(Feb. 24, 2005)**

The staff should review the generic potential for boron concentration and precipitation to interfere with core cooling following a loss-of-coolant accident (LOCA). **(Feb. 24, 2005)**

Our discussions also revealed that there is not a good understanding of the deposition of boron on the overheated portions of the fuel rods, which are predicted to be exposed for up to 45 minutes during some small-break LOCAs. Splashes and droplets of borated water may be deposited on the exposed fuel rods and spacer grids and the water will evaporate, leaving boric acid deposits that will decompose at the prevailing temperature to form dry boric oxide. We encourage the staff to establish a basis for a quantitative assessment of these phenomena as it considers the potential for boron concentration and precipitation to interfere with core cooling following a LOCA. **(Feb. 24, 2005)**

VERMONT YANKEE NUCLEAR POWER STATION (VY)

The ACRS has historically opposed a general granting of containment overpressure credit. In determining whether such credit should be granted, one aspect to be considered is whether practical alternatives exist, such as the replacement of pumps with those with less restrictive NPSH requirements. If no practical alternatives are available, important considerations include (1) the length of time for which containment pressure credit is required and (2) the margin between the magnitude of the pressure increment that is being granted and the expected minimum containment pressure. Another consideration is the nature of the containment design and whether it provides a positive indication of integrity, prior to the event, as is the case in subatmospheric and inerted designs.
(Jan. 4, 2006)

Under the EPU conditions at VY, the general design requirements regarding single failures in design-basis accidents do not prevent granting of the overpressure credit for the LOCA scenario of concern. The worst single failure that was identified by the licensee involves loss of one train of heat removal from the suppression pool. Conservative, bounding calculations show that the containment overpressures during this scenario are higher than that needed to provide sufficient NPSH. Allowing no credit for containment overpressure is equivalent to assuming an additional failure that causes loss of the overpressure. Thus, for all scenarios involving only a single failure, sufficient NPSH is available to ensure that pump cavitation damage is avoided. To maintain defense-in-depth, however, it has been staff practice to require the assumption that containment overpressure is not available in assessing the potential for pump damage. **(Jan. 4, 2006)**

Design-basis accidents are typically analyzed using conservative methodologies and input assumptions to ensure safety in spite of uncertainties in input and methodology. An alternative approach is to use realistic analyses with a more complete and explicit consideration of uncertainties. Such a methodology has not yet been fully developed for analysis of the need for containment overpressure credit. The staff and the licensee have instead performed sensitivity analyses to determine the effect of relaxing some of the conservative assumptions. More realistic values were used for a number of input parameters to determine the associated reduction in the predicted temperature of the suppression pool, which is the major parameter in determining whether overpressure credit is necessary. The

staff concluded that, on a more realistic but still conservative basis, the temperature of the suppression pool would not become high enough in the LOCA scenario to require a credit for containment overpressure. **(Jan. 4, 2006)**

Independent risk analyses were performed by the staff and the licensee to determine the potential risk significance of granting credit for containment overpressure. These analyses included the conservative assumption that the emergency core cooling system (ECCS) success criteria would not be met whenever containment overpressure is lost and design-basis analyses would suggest that overpressure credit was needed, although the licensee's sensitivity studies indicated that peak suppression pool temperature would probably not be high enough that containment overpressure credit would be required. The results of the analyses indicate that the overall risk associated with the EPU is small and that the change in risk resulting from allowing the requested containment overpressure credit is also small. **(Jan. 4, 2006)**

Although we concur with the staff's conclusion to grant credit for containment overpressure, we would have preferred to see the assessment performed and presented in a more coherent manner, with a more complete and rigorous consideration of uncertainties. The staff is developing additional guidance to be used in the consideration of overpressure credit in the future. We look forward to reviewing their proposed approach. **(Jan. 4, 2006)**

BEAVER VALLEY POWER STATION (BEAVER VALLEY), UNITS 1 AND 2

In our report of February 24, 2005 related to the Waterford 3 uprate, we indicated the need for the staff to develop a better understanding of the properties of highly concentrated boric acid in a boiling system. A more detailed treatment of the thermal-hydraulic conditions within the core region is needed to better define the conditions leading to recirculation and mixing within the vessel and lower plenum. In its response to our letter, the staff stated that this issue should be addressed by the industry as part of satisfying the long-term cooling requirements of 10 CFR 50.46. We look forward to reviewing progress on this issue. **(May 22, 2006)**

For Unit 1, containment overpressure credit has been granted by the staff to provide net positive suction head for the containment spray pumps that recirculate coolant from the containment sump. Containment spray flow through heat exchangers provides long-term removal of heat during a LOCA. The duration of time for which overpressure credit is required is less than 20 minutes. FENOC provided results from tests performed on this pump design that demonstrate an ability to operate for this period without damage. Under EPU conditions, the amount of overpressure and duration of credit required are only slightly increased. We concur with the staff's decision to grant overpressure credit under these conditions. Because of a difference in the location of the pumps in Unit 2, no overpressure credit is required. **(May 22, 2006)**

The power uprate will lead to additional fluence and embrittlement of the reactor vessel at the end of life for the two units. Based on results obtained from surveillance capsules, FENOC has estimated the shift in the pressurized thermal shock reference temperature (RTPTS) at the end of extended life. These estimates have been independently confirmed by the staff. The final value of RTPTS for each

vessel is less than the pressurized thermal shock screening criterion of 270oF. The upper shelf energies exceed 50 ft-lbs. We conclude that radiation-induced vessel embrittlement is a manageable issue at the power uprate conditions. **(May 22, 2006)**

FENOC has performed a systematic assessment of components for which vibration could be induced by higher velocities following the power uprates. The main steam condenser at Unit 2 will be staked; the Unit 1 condenser was staked previously. There is extensive industry operating experience with the steam generators in use at both units for the conditions that will be encountered at Beaver Valley without any indication of vibration-induced failures. The steam dryers in these units are subject to much lower flow velocities than those in boiling water reactors for which flow-induced vibrations have been a power uprate issue. FENOC has committed to performing pre-EPU and post-EPU walkdowns to identify vibration issues should they occur. **(May 22, 2006)**

R.E. GINNA NUCLEAR POWER PLANT (GINNA)

full spectrum of loss-of-coolant accident (LOCA) events has been analyzed at the uprated power. The results of these analyses show substantial margin to the established regulatory limits on peak clad temperature, oxidation, and hydrogen generation. The emergency core cooling system configuration at Ginna is somewhat different from later plant designs. The high capacity, low-pressure system injects through two lines directly into the upper plenum. The high-pressure system also has high capacity and the accumulators inject at a relatively high pressure of 700 psia. This configuration of systems is quite effective in providing cooling over the entire spectrum of breaks. **(May 22, 2006)**

At the time at which recirculation is initiated in a large LOCA, the sump temperature is too high to meet the net positive suction head limits for the high-pressure injection system. Thus, when recirculation is initiated, the low-pressure upper plenum injection system is switched from the injection mode to the recirculation mode but the high-pressure injection system is turned off. For a hot-leg break, there is some concern that, with injection occurring only on the hot side of the core, emergency core cooling water could escape out the break without effectively mixing in the core. Boric acid could concentrate within the vessel and potentially deposit within the core region. The licensee has performed analyses to determine when cold-leg injection should be reinitiated to flush the system and ensure that the concentration of boric acid does not approach saturation. The emergency operating procedures have been modified accordingly. **(May 22, 2006)**

The potential for flow-induced vibration associated with higher secondary side flow rates has been assessed for the steam generators, feedwater heaters, condenser tubes, and moisture separator reheaters. Within the vibration monitoring program, a baseline will be established by a walkdown prior to EPU. After EPU plant modifications have been made, walkdowns will be performed at the initial power level and at the uprated power level.

Because the temperatures in the primary system will be somewhat higher after EPU, we requested that the licensee identify those components that contain Alloy 600 and its associated weld materials (Alloy 82/182) for which increased stress

corrosion cracking might be expected. The licensee explained that these components are all located in regions of the primary system that will not experience high temperatures or are not load bearing. **(May 22, 2006)**

BROWNS FERRY NUCLEAR PLANT, UNIT 1, 5-PERCENT POWER UPRATE

…. because all of the information necessary to support the review of the EPU is not yet available, the licensee has requested a two-step approach. This approach involves, as a first step, an interim approval of a 5-percent power uprate for Unit 1 to raise its power to the same level and operating conditions as Units 2 and 3. The second step will involve a 15-percent EPU for all three units later this year. **[February 16, 2007]**

Granting of containment overpressure credit during long-term loss-of-coolant accident (LOCA) and 10 CFR Part 50 Appendix R fire scenarios at 120-percent of the original licensed thermal power (OLTP) will require support by more complete evaluations. **[February 16, 2007]**

In determining whether credit for containment overpressure should be granted, we have noted in previous reports a number of important considerations. They include whether practical alternatives exist, such as the replacement of pumps with new pumps with less restrictive NPSH requirements; whether the containment design provides a positive indication of integrity before the event, as is the case in inerted containments; and the length of time for which containment pressure credit is required and the margin between the containment pressure required and the expected minimum containment pressure. The ultimate consideration is the risk significance of granting credit for containment overpressure. **[February 16, 2007]**

Because of the plant configuration, the extent of modification required, and the worker dose that would be involved, we conclude that there are no practical design modifications that would preclude the need to consider the request for containment overpressure credit for most of the scenarios. However, for the Appendix R scenario, protecting a second RHR pump would eliminate the need for the credit and may be a feasible alternative. **[February 16, 2007]**

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 AND 2 EXTENDED POWER UPRATE APPLICATION

An appropriate margin should be added to the operating limit minimum critical power ratio (OLMCPR) as an interim measure to account for uncertainties in the void fraction correlation and the lack of data for its validation at void fraction above 90 percent. This interim measure should be reviewed when PPL submits more detailed analyses that account for the effect of uncertainties in the void fraction on the OLMCPR. **[December 20, 2007]**

The staff should develop the capability and perform a thorough review and assessment of the risk of pellet-cladding interaction (PCI) fuel failures with conventional fuel cladding, during anticipated operational occurrences (AOOs. **[December 20, 2007]**

Review Standard (RS)-001, "Review Standard for Extended Power Uprates," provides a structured process for the review of EPU applications. The guidance document should be improved to include cross-referencing of related sections between the power uprate safety analysis report (PUSAR) and the staff's SEs. **[December 20, 2007]**

At the high-power/low-flow conditions that are susceptible to instabilities, the presence of bypass voiding in the upper regions of the core leads to errors in the local power range monitor (LPRM) signals. SSES is an Option III plant, which relies on OPRMs to initiate scram. The LPRM signals feed into the OPRMs, and therefore the errors in the LPRM signals will also affect the OPRMs. We concur with the staff that errors in the LPRM signals caused by bypass voiding must be accounted for in the determination of the OPRM setpoint. The methodology proposed for determining OPRM setpoint values is acceptable. **[December 20, 2007]**

The January 17, 2008, [EDO] response provides the staff's basis for concluding that additional OLMCPR margin was not necessary to ensure adequate safety. We are not persuaded by the staff's arguments. To clarify the matter, we have reexamined the influence of uncertainties in void fraction prediction on OLMCPR based on analyses conducted by our consultants. …..These analyses as well as other assessments indicate that uncertainties in void fraction prediction can result in a significant change in CPR. **[June 3, 2008]**

We continue to recommend that in view of the many more fuel assemblies that will operate near limiting conditions for the EPU operation, the effects of the uncertainties in void fraction predictions on the OLMCPR be assessed over their full ranges. **[June 3, 2008]**

HOPE CREEK GENERATING STATION EXTENDED POWER UPRATE APPLICATION

Monitoring during power ascension testing will provide reasonable assurance that unanticipated vibration modes induced in the steam dryer will be detected, should they occur. **[May 2, 2008]**

The acoustic coupling model has limited validation; however, its use is acceptable in the HCGS EPU application because of the predicted large stress margin of the steam dryer. **[May 2, 2008]**

We concur with the staff that large-transient tests, such as main steam isolation valve (MSIV) closure or generator load rejection, that would result in a reactor trip should not be required. **[May 2, 2008]**

STRETCH POWER UPRATE APPLICATION FOR THE MILLSTONE POWER STATION, UNIT 3

During the recirculation phase, containment recirculation spray pumps and the ECCS pumps take suction from the containment sump. MPS3 has already

modified its sump strainers to address the concerns of Generic Safety Issue - 191. Net positive suction head calculations show that containment overpressure credit is not needed for any design-basis accident. **[July 23, 2008]**

The licensee has submitted an acceptable application for SPU [Stretch Power Uprate] condition, including a revised supplemental environmental report, proposed technical specification changes, and changes to the associated bases, proposed instrument and control changes (including updating the simulator), and changes to the operator training program. The staff's review was thorough and complete. Based on our review of the matters associated with this application, we conclude that DNC [Dominion Nuclear Connecticut, Inc.] application for MPS3 [Millstone Power Station, Unit 3] SPU should be approved. **[July 23, 2008]**

5. INTERNATIONAL PERSPECTIVES ON POWER UPRATE

The greater demand for electricity and the available capacity in safety margins in some of the operating nuclear power plants have also prompted nuclear utilities in other parts of the world to request license modifications to enable operation at a higher power level, beyond the provisions of the original license [10]. Examples of a successful power increase can be found among different types of reactors, such as PWRs, BWRs, VVERs and others. The Loviisa NPP in Finland, for instance, increased thermal power by 9.1% between the years of 1998 and 2000 [11]. Table 2 provides a summary of status of power uprates in other countries.

Much experience has been gained in Belgium on power uprates of nuclear power plants. Out of the seven Belgian nuclear units in operation, power uprates have been implemented for three of them (Doel 3, Tihange 1, and Tihange 2), while a power uprate is under way for a fourth plant (Doel 2). For Tihange 2, the power uprate was implemented in two steps of about 5 %. For Doel 3, Tihange 1, and that planned for Doel 2, the single-step power uprate is also coupled with a steam generator replacement. To allow a final uprate value of 10%, core design evolutions, major equipment modifications and changes of instrumentation setpoints were needed. Also, new methodologies were introduced to take advantage of unnecessarily large safety margins in some safety analyses [11].

France has 59 nuclear reactors operated by Electricite de France (EdF). All French units are PWRs of three standard types designed by Framatome (now Areva NP): 900 MWe (34), 1300 MWe (20) and 1450 MWe N4 type (4). The first two derived from US Westinghouse types. In light of operating experience, EdF uprated its four Chooz and Civaux N4 reactors from 1455 to 1500 MWe each in 2003. Over 2008-2010, EdF plans to uprate five of its 900 MWe reactors by 3%. Then in 2007 EdF announced that the twenty 1300 MWe reactors would be uprated some 7% within existing license limits [12].

A gradual increase in reactor thermal power began in the German pressurized water plants of the 1300 MW series more than a decade ago. In this way o a power uprate of approximately 5% of the original nominal power has been implemented. Examples of German PWRs with such uprates are Philippsburg 2, Emsland, Isar 2, and Unterweser [11].

Spain is notable for power plant uprates. It has a program to add 810 MWe (11%) to its nuclear capacity through upgrading its nine reactors by up to 13%. Some 519 MWe of the overall increase is already in place. Cofrentes was uprated 2% in 1988, another 2.2% in 1998, 5.6% in 2002 and 1.9% in 2003, taking it to 112% of original capacity. Tentative plans will take it to 120% later in the decade [12].

During the eighties, 7 out of 8 BWRs in Sweden were uprated between 5.9 and 10.1 %. One of the PWRs was uprated as well. Most of the Swedish reactors are planning further uprates in the coming years; a few of them have already been given a first approval by the Swedish government and the regulatory body. The government is working with the utilities to expand nuclear capacity to replace the 1200 MWe lost in

closure of Barsebäck 1 & 2. By the end of 2008, some 1050 MWe had been added to the ten surviving reactors [12]. More recently, the Swedish Radiation Safety Authority (SSM) has given its approval for test operation of OKG's Oskarshamn 3 nuclear power reactor following a power uprate of some 20% from 1200 MWe to 1450 MWe net. The trial operation of the reactor at the higher power output can continue for one year. OKG must then submit another application for routine operation at the higher level [13].

In Switzerland, three utilities have requested and received regulatory authorization for power uprates. The Gösgen plant was permitted to undergo a 6.9% power uprate in 1985. In 1992 the Mühleberg power plant also received permission for a power uprate of about 10%. On the other hand, the Leibstadt power plant twice requested and received permission to uprate. This included an uprate of 4.2% in 1985 and, subsequently in 1998, the plant was permitted to uprate by an additional 14.7% [12]

In Korea the first power uprating projects are ongoing for 4 units out of 20 operating ones. The two affected plants are Kori Units 3&4 and Yongwang Units 1&2, which are PWR-type reactors. The NSSS supplier was Westinghouse and the original electrical output was 950MWe. Uprating will result in the thermal power increase from 2775 MWth to 2900 MWth (4.5%) [11].

In October 2003, the IAEA, in cooperation with the OECD/NEA, organized a meeting to provide an international forum for presentation and discussion on topics related to the impact of power uprates on plant safety margins [10]. It was concluded that the regulatory practice varies from country to country and obtaining a consensus between different countries is not easy. The regulatory positions were broadly characterized as two categories:

- The current acceptance criteria and absolute safety margins should be preserved
- Current acceptance criteria should be fulfilled; licensing margins are allowed to become smaller.

It was suggested that an integration of deterministic and probabilistic approach could provide the basis for decision-making if modifications are acceptable; permission is only granted when these goals are fulfilled:

- Acceptance criteria are met.
- Licensing margins and/ or safety analysis are acceptable.

Another approach put forward by several countries during the Technical Meeting was the acceptance of a limited risk increase in the short term, while risk in the long term is continuously decreased. One of the reasons to address the safety and licensing margins is the effect of cumulative changes in the plants. Small changes that do not by themselves warrant an in-depth safety review may accumulate over the years and the plant conditions may prove to be outside the scope of the safety analysis report. Integral assessment of the impact of all changes was recommended [10].

Table 2 Status of Power Uprates in other Countries

Country	Status of Power Uprates
Belgium	Power uprate have been implemented for 4 out of 7 plants in operation Doel 2 (10%) Doel 3 (10%) Tihange 1 (8%) Tihange 2 (two steps of about 5%)
Finland	Loviisa (9.1%)
France	Four N4 reactors (3%) Five 900 MWe reactors (3%) Twenty 1300 MWe reactors (7%)
Germany	A gradual increase in reactor thermal power of the 1300 MW PWR series Philippsburg 2 (5% in two steps) Emsland (5% in 2 steps) Isar 2 (5% in 2 steps) Unterweser (4.5%)
Spain	Has a program to add 810 MWe (11%) to its nuclear capacity through upgrading its nine reactors by up to 13%
Sweden	BWRs Oskarshamn 1 Oskarshamn 2 Oskarshamn 3 (trial operation at 20% uprate) Ringhals 1 (3.4%) Forsmark 1 (8.6%) Forsmark 2 (8.5%) Forsmark 3 (15.8%) PWRs Ringhals 2 Ringhals 3 Ringhals 4
Switzerland	Gösgen (6.9% in 1985) Mühleberg (10% in 1992) Leibstadt (4.2% in 1985 and 14.7% in 1998)
Republic of Korea	Power uprate are being implemented for 4 units out of 20 operating ones Kori Units 3&4 (4.5%) Yongwang Units 1&2 (4.5%)

6 SUMMARY AND CONCLUSIONS

An overview of power uprate regulations and the role of ACRS in power uprate review process were presented. The ACRS has contributed significantly to the success of the power uprate program by establishing expectations on the quality of the power uprate license amendment requests and supporting documentations. ACRS was instrumental in the staff development of a review standard for extended power uprates.

Since 1998, ACRS has reviewed fifteen applications for power uprates. The ACRS recommendations, observations, and comments provided during its past reviews of power uprate license amendment requests were presented to provide insights and perspectives on previous Committee's review of power uprate applications.

An overview of international perspectives on power uprates was also presented.

7 REFERENCES

1. *Code of Federal regulations,* Title 10, part 50, *Domestic Licensing of Production and utilization Facilities.*

2. U.S. Nuclear Regulatory Commission, “Review Standard for Extended Power Uprates,” NRR, RS-001, Revision 0, December 2003.

3. U.S. Nuclear Regulatory Commission, Power Uprates, http://www.nrc.gov/reactors/operating/licensing/power-uprates.html

4. U.S. Nuclear Regulatory Commission, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," NUREG-0800.

5. U.S. Nuclear Regulatory Commission, “Staff Requirements Memorandum, COMSAJ-97-008 – Discussion on Safety and Compliance,” August 25, 1997.

6. U.S. Nuclear Regulatory Commission, “Guidance on Risk-Informed Decisionmaking in License Amendment Reviews,” RIS-2001-002, January 18, 2001.

7. U.S. Nuclear Regulatory Commission, “Staff Requirements Memorandum, Meeting with Advisory Committee on Reactor Safeguards (ACRS),” December 20 2001.

8. U.S. Nuclear Regulatory Commission,” Review of ACRS Recommendation for the Staff to Develop a Standard Review Plan for Power Uprate Reviews,” SECY-02-0106, June 14, 2002.

9. *GE Nuclear Energy, Topical Report, NEDC-32424P-A, "Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate," (ELTR-1) February 1999 (Proprietary).*

10. International Atomic Energy Agency, “Implications of power uprates on safety margins of nuclear power plants,” Report of a technical meeting organized in cooperation with the OECD/NEA held in Vienna, 13–15 October 2003, IAEA-TECDOC-1418, September 2004.

11. International Atomic Energy Agency, “The Role of I&C Systems in Power Uprating Projects in Nuclear Power Plants,” Report prepared within the framework of the Technical Working Group on Nuclear Power Plant Control and Instrumentation, IAEA-TECDOC-XXXX, Draft, May 2007.

12. World Nuclear Association, http://www.world-nuclear.org/info/inf42.html

13. World Nuclear News, October 2, 2009
http://www.world-nuclear-news.org/newsarticle.aspx?id=26229