U.S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation

NRR OFFICE INSTRUCTION

Change Notice

Office Instruction No.:	**LIC-112**
Office Instruction Title:	**Power Uprate Process**
Effective Date:	**February 17, 2009**
Approved By:	**James T. Wiggins**
Date Approved:	**February 12, 2009**
Primary Contact:	**Thomas Alexion** **301-415-1326** **Thomas.Alexion@nrc.gov**
Responsible Organization:	**NRR/DPR/PGCB**

Summary of Changes: This is the initial issuance of Office Instruction (OI) LIC-112, "Power Uprate Process."

Training: E-mail announcement with recommended self-study for staff involved with power uprates.

ADAMS Accession No.: ML082210335

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OFFICE	DPR/PGCB/PM	DPR/PGCB/LA	DPR/PCGB/BC	DSS/D	DCI/D	DE/D
NAME	TAlexion	CHawes	MMurphy	WRuland	MEvans	PHiland
DATE	08/07/08	08/08/08	12/09/08	10/16/08	10/14/08	10/21/08
OFFICE	DRA/D	DLR/D	DORL/D	DIRS/D	DPR/D	OGC (nlo)
NAME	MCunningham	BHolian (SSLee for)	JGiitter	FBrown (MCheok for)	TMcGinty	DRoth
DATE	10/31/08	09/22/08	10/14/08	10/23/08	01/08/09	10/10/08
OFFICE	PMDA/D	NRR/ADES	NRR/ADRO	NRR/D		
NAME	MGivvines BFicks/f/	JGrobe	BBoger	ELeeds JWiggins /f/		
DATE	02/11/09	02/09/09	02/10/09	02/12/09		

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NRR OFFICE INSTRUCTION
(LIC-112)

Power Uprate Process

1. POLICY

The Commission determined that applications for power uprates should be assigned high priority and should be conducted in the most effective and efficient manner (Staff Requirements - COMNJD-01-0001 – Power Uprate Applications, dated May 24, 2001, ML011440274).

2. OBJECTIVES

It is the objective of this office instruction to strengthen the coordination of all aspects of power uprate activities and identify roles and responsibilities for headquarters and regional points of contact for power uprates. This office instruction addresses several of the Office of Inspector General's (OIG's) recommendations in the OIG Audit Report, "Audit of NRC's Power Uprate Program" (OIG-08-A-09), dated March 28, 2008.

3. BACKGROUND

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. This power level is included in the license and technical specifications for the plant. NRC controls any change to a license or technical specification, and the licensee may only change these documents after NRC approves the licensee's application for change. The process of increasing the maximum power level at which a commercial nuclear power plant may operate is called a power uprate.

Improvements in instrument accuracy, computational tools and engineering models, in addition to plant hardware modifications, have allowed licensees to request power uprates while maintaining safety margins. The three categories of power uprates are:

- measurement uncertainty recapture (MUR) power uprates
- stretch power uprates (SPU)
- extended power uprates (EPU)

MUR power uprates are less than 2 percent above the current licensed thermal power (CLTP) limit and are achieved by implementing enhanced techniques for calculating reactor power. This involves the use of state-of- the-art feedwater flow measurement devices to more precisely measure feedwater flow, which is used to calculate reactor power. More precise measurements reduce the degree of uncertainty in the power level that licensees are required to assume when performing emergency core cooling system analyses, which allows licensees to propose an increase in the CLTP limit.

SPUs are typically up to 7 percent above the original licensed thermal power (OLTP) limit and are within the original design capacity of the plant. The actual value for percentage increase in power a plant can achieve and stay within the SPU category is plant-specific and depends on the operating margins included in the original design of a particular plant. SPUs usually involve changes to instrumentation setpoints but do not involve major plant modifications.

EPUs have been approved for power increases as high as 20 percent above the OLTP limit. These uprates require significant modifications to major balance-of-plant equipment such as the high pressure turbines, condensate pumps and motors, main generators, and/or transformers.

The convention for specifying the percent uprate in an individual power uprate application is that the application should be quantified in terms of the percent uprate from the CLTP, with an additional statement designating the total increase from the OLTP. For example, on Month dd, 2008, the licensee for Plant ABC requested a 6.4 percent EPU from the CLTP, which equates to about a 14 percent uprate from OLTP due to NRC's approval of a 7.4 percent EPU for Plant ABC in 1993.

MUR power uprates, SPUs, and EPUs may be approved in steps. However, there typically are limits to the percent uprate for MUR power uprates and SPUs. There are no limits for EPUs, provided the licensee's technical analyses can support the EPU and the NRC staff approves it.

The available technology for ultrasonic flow meters currently supports MUR power uprates up to about 1.7 percent.

The staff interprets the phrase "the operating margins included in the original design of a particular plant" in the SPU definition to mean the "operating margins included in the design of a particular plant at the OLTP." For example, a plant could receive a 3 percent SPU and a 4 percent SPU at two different times, as long as the plant remained within the operating margins included in the design capacity of the plant at the OLTP.

MUR power uprates can be approved before SPUs and/or EPUs, or after SPUs and/or EPUs. This is facilitated by the fact that the emergency core cooling system (ECCS) analyses supporting MUR power uprates are generally the same ECCS analyses that were performed by licensees, and reviewed and approved by NRC, at the pre-MUR power level (i.e., 102 percent of the CLTP value in effect just before the MUR power uprate).

The following table provides examples of plants with multiple NRC-approved power uprates, with the uprate percentages given in terms of the CLTP limits:

Licensee	MUR Power Uprate Percent (and Yr)	SPU Percent (and Yr)	EPU Percent (and Yr)
Hatch 1	1.5 (2003)	5 (1995)	8 (1998)
Hatch 2	1.5 (2003)	5 (1995)	8 (1998)
Susquehanna 1	1.4 (2001)	4.5 (1995)	13 (2008)
Susquehanna 2	1.4 (2001)	4.5 (1994)	13 (2008)

Guidance for MUR power uprates is provided in Regulatory Issue Summary (RIS) 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power

Uprate Applications," dated January 31, 2002 (ML013530183) and in RIS 2007-24, "NRC Staff Position on Use of the Westinghouse Crossflow Ultrasonic Flow Meter for Power Uprate or Power Recovery," dated September 27, 2007 (ML063450261). Guidance for EPUs is provided in Review Standard (RS)-001, Revision 0, "Review Standard for Extended Power Uprates," dated December 2003 (ML033640024). There is no specific guidance for SPUs. The staff should use previously approved SPUs, along with RS-001, for guidance.

4. BASIC REQUIREMENTS

Power uprate requests are submitted to NRC as license amendment requests. This regulatory process is governed by 10 CFR 50.90, 50.91 and 50.92, and provides for the amending of commercial nuclear power plant licenses and technical specifications related to power uprates. It is the same regulatory process used for other types of amendments. NRR Office Instruction LIC-101, "License Amendment Review Procedures," provides guidance for processing license amendment applications. Therefore, this office instruction focuses on detailed staff guidance that is unique to processing power uprate applications.

5. RESPONSIBILITIES AND AUTHORITIES

Division of Policy and Rulemaking (DPR)

The Generic Communications and Power Uprate Branch (PGCB) is the coordinating agent for power uprate activities. PGCB has a Lead Project Manager (Lead PM) for power uprates. The Lead PM provides oversight, information and/or guidance to internal and external stakeholders regarding approved, pending, and expected power uprate applications. The Lead PM is responsible for providing an annual power uprate status report to the Commissioners and providing high-level briefings or briefing materials on power uprates to NRC senior management (e.g., background information regarding power uprates previously approved, status of power uprates under review, and challenges to the timely review of current and future power uprate applications). The Lead PM is responsible for updating NRC's internal and external guidance on the power uprate review process if needed (e.g., in a Generic Communication that references the updated guidance document), briefing external stakeholders on power uprates, initiating the semi-annual survey of licensees regarding their future plans for power uprate applications, compiling the results of the survey in a document (e.g., table and/or chart) that includes current power uprate applications under review along with future applications expected, and maintaining NRC's public power uprate website.

The Lead PM maintains the generic review schedules for the three types of power uprates. The generic review schedules include standard interim milestones [e.g., completion of acceptance reviews, preparation of requests for additional information (RAIs), providing safety evaluation (SE) inputs] and they are shown in Appendix D. The generic review schedules help the Plant Project Managers (Plant PMs) establish the initial plant-specific review schedule for each power uprate application. The Lead PM provides information and guidance to the Plant PMs on questions or significant problems relating to power uprate reviews. The Lead PM is responsible for ensuring the preparation of an Executive Director for Operations (EDO) Daily Note or an EDO Weekly Highlight (1) when a power uprate application is received by NRC, (2) at the conclusion of the acceptance review process, and (3) when the amendment review is completed (either approved or denied) or withdrawn. The Lead PM is responsible for ensuring that the Regional power uprate point-of-contact is informed when a power uprate application has been accepted by NRC for detailed technical review, so that the Region can begin considering any inspection activities that need to take place before the power uprate is approved and implemented (as discussed later in this office instruction).

On an annual frequency, the PGCB Branch Chief will solicit inputs, via memorandum to the applicable Branch Chiefs, from internal NRC stakeholders on significant new information, trends, best practices, and lessons learned related to power uprate reviews. Responses should be provided by memorandum from the responding Branch Chief to the PGCB Branch Chief. In addition, the Lead PM will accept inputs from internal stakeholders at any time if the stakeholder desires to provide the information promptly, before it is forgotten or so that it can be shared quickly. An e-mail distribution group called “Power_Uprate Distribution” is being created for the purpose of providing inputs at any time. (Applicable NRR and Regional Branch Chiefs and their staffs are currently being added to this distribution group. In the interim period, e-mails can be sent directly to the Lead PM while the new distribution group is being made available to all applicable staff.) All e-mail inputs should be approved by the appropriate Branch Chief before being e-mailed and any information provided via e-mail should be resubmitted in writing in response to the annual solicitation. On an annual frequency, this information will be reviewed and combined by the Lead PM, and approved at the appropriate management level for dissemination to the internal stakeholders to ensure knowledge transfer.

The Lead PM provides estimates of resource needs for current and future power uprate reviews to support information needs for NRR’s budget. The Lead PM maintains the table of resource assumptions used for modeling power uprate reviews as shown in Appendix C. The Lead PM provides long-range forecasting (tentative schedules) to the Advisory Committee on Reactor Safeguards (ACRS) staff for briefing future EPU reviews to the ACRS subcommittee and full committee. The Lead PM ensures that the Division of Inspection and Regional Support maintains power uprate point-of-contact(s) for each of the Regions.

The PGCB Branch Chief and/or DPR management (or others as determined by DPR management) will provide just-in-time SE refresher training as part of the kick-off meeting for EPUs. The kick-off meeting is the initial meeting typically with the power uprate lead project manager, the plant project manager, and the technical reviewers assigned to the EPU review, once an EPU application has been accepted by NRC. At the kick-off meeting, the staff will discuss the review schedule, any plant-specific issues, recent experience with EPU reviews, and SE inputs. The SE training will address the purpose and content of SE inputs.

The PGCB Branch Chief and DPR management provide oversight to ensure that the Lead PM is performing the duties discussed above and that the power uprate performance measures (i.e., review timeliness goals) discussed later in this office instruction are being met or there is adequate justification for not meeting them. The Director, DPR, is responsible for overall implementation of this office instruction and power uprate activities.

Division of Operating Reactor Licensing (DORL)

DORL conducts the semi-annual survey of licensees regarding their future plans for submitting power uprate applications. Based on this survey, starting about one year prior to the submittal of the application, the Plant PM should solicit pre-application interactions (e.g., meetings, telephone calls, review of draft submittals) between the licensee, the technical staff, and the Lead PM to discuss the scope of the power uprate application and ensure that challenges and success paths related to previous reviews are understood and addressed in the forthcoming application. (Previously approved power uprates along with the NRC's supporting SEs, can be found in the NRC's public power uprate website at http://www.nrc.gov/reactors/operating/licensing/power-uprates/approved-applications.html). The Plant PM should encourage the licensee to focus the discussions on those items that are new, complex or different as compared to previously approved power uprates. The Plant PM should invite the appropriate technical staff and the Lead PM to the pre-application interactions. These interactions are of great importance for EPUs.

The Plant PM is responsible for establishing the initial plant-specific power uprate review schedule for a power uprate application, in consultation with the Lead PM. The Plant PM should consult with the Lead PM for information and guidance on questions or significant problems relating to power uprate reviews, and inform the Lead PM on delays in the review schedule. The Plant PM is responsible for providing review schedule updates to the Lead PM regarding their plant-specific power uprate applications under review. Typical schedule information includes the projected completion dates of obtaining all SE inputs from the technical staff and the projected amendment review completion date. The Plant PM must coordinate/communicate with the Lead PM on all schedule issues.

The Plant PM is responsible for conducting, coordinating and managing the NRC's review of a power uprate license amendment application just like any other license amendment application, per LIC-101. The Plant PM is responsible for briefing NRC senior management on the status of an individual power uprate application, if requested. The Plant PM should coordinate the acceptance review in accordance with LIC-109, "Acceptance Review Procedures." Typical problem areas with accepting previous power uprate applications include linked amendments and incomplete applications. LIC-109 explains these and other acceptance review criteria which should be thoroughly considered when performing the acceptance reviews. Due to the high visibility of power uprate reviews, the Plant PM should document the results of the staff's acceptance review in a letter(s) to the licensee.

The Plant PM should coordinate the power uprate review in accordance with NRR Office Instruction COM-109, "NRR Interfaces With the Office of the General Counsel." This ensures that appropriate legal advice is received in order to assure that official actions taken by NRR staff are in accordance with the laws of the United States. Coordination with the Office of the General Counsel (OGC) is especially important if a hearing is requested regarding the power uprate application.

The Plant PM is responsible for ensuring that all needed SE inputs are being prepared by the appropriate technical staff for inclusion in the final, combined SE that is issued with the license amendment. For MUR power uprate applications, some of the technical branches may decline providing SE input and indicate that they only need to concur on the outgoing license amendment that approves the uprate. In these cases, the Plant PM performs the technical review and provides the SE input. The Plant PM's review should consist of finding that the licensee's application has addressed the appropriate technical areas in RIS 2002-03, Attachment 1, and that for each area the licensee determined its existing analysis of record is bounding for the MUR power uprate. If the licensee provides something other than a bounding analysis to address a technical area (e.g., the licensee revised their analysis with revised assumptions and/or methods), the technical branch should perform the detailed review of the application and provide SE input.

The Plant PM is responsible for providing the draft combined SE to the ACRS staff for proposed power uprates greater than 7 percent above the OLTP limit (excluding proposed MUR power uprates),[1] and for other power uprate reviews that involve important changes to the plant or present novel issues, the review of which might benefit from ACRS participation.[2] Generally, the draft SE is transmitted by memorandum from DORL to the ACRS at least one month prior to the ACRS subcommittee meeting. The Plant PM should provide 15 electronic copies of the draft SE with the memorandum. The Plant PM should also provide 15 electronic copies of the licensee's supplemental responses. The memorandum should include a table that provides cross-references between the staff's numbering of the specific technical review areas in the SE (e.g., the EPU Review Standard RS-001 numbering scheme) and the applicable sections of the licensee's numbering scheme and the licensee's supplemental responses. The Plant PM is responsible for providing this table.

The Plant PM coordinates the briefings to the ACRS subcommittee and full committee. The Plant PM provides comments on the draft ACRS subcommittee agenda provided by the ACRS staff engineer. The Plant PM notifies the technical staff and the licensee once the ACRS staff engineer provides the final ACRS subcommittee agenda. The Plant PM contacts the ACRS staff member responsible for power uprates for any specific guidance in preparing for the briefings. Electronic slides (e.g., Microsoft PowerPoint presentation) are usually presented by the Plant PM and selected technical staff reviewers. The Plant PM provides the list of attendees with their company and country of origin to the ACRS staff engineer for the ACRS subcommittee and ACRS full committee meetings, so the ACRS staff engineer can enter the attendees into the visitor access request system.

At the conclusion of the ACRS subcommittee meeting, the ACRS Subcommittee Chairman will notify the Plant PM, the staff, and the licensee if the power uprate is technically sufficient to be presented to the ACRS full committee. If the power uprate is not technically sufficient to be presented to the ACRS full committee, the ACRS Subcommittee Chairman will explain to the Plant PM, the staff, and the licensee which topic areas need to be presented at another ACRS subcommittee meeting. If the power uprate is technically sufficient to be presented to the ACRS full committee, the ACRS Subcommittee Chairman will tell the Plant PM, the staff, and the licensee which topic

[1] See memorandum from R. W. Borchardt, Executive Director for Operations, to Frank P. Gillsepie, Executive Director, Advisory Committee on Reactor Safeguards, Subject: Advisory Committee on Reactor Safeguards (ACRS) Review of Power Uprates, dated June 23, 2008 (ML081410658).
[2] See memorandum from John T. Larkins, Executive Director, Advisory Committee on Reactor Safeguards, to James E. Dyer, Director, Office of Nuclear Reactor Regulation, Subject: Kewaunee Nuclear Power Plant – Advisory Committee on Reactor Safety Review of Stretch Power Uprate Amendment (TAC No. MB9031), dated October 9, 2003 (ML040620143).

areas need to be presented at the ACRS full committee meeting. Following the ACRS full committee meeting, the ACRS typically writes a letter (with conclusions and recommendations) to the NRC Chairman regarding the power uprate application. The Office of the Executive Director for Operations then tasks the NRR staff with responding to the ACRS with a Green Ticket. The Plant PM prepares the response to the ACRS and solicits input from the technical staff and/or the Lead PM to address any technical and/or process issues.

If appropriate, the Plant PM should begin drafting a Communication Plan (typically only needed for EPUs) about four to six weeks prior to the projected amendment review completion date, with review completion meaning either NRC approval or denial. This Communication Plan should be developed in accordance with COM-201, “Public Outreach and Communication Plans.” The Plant PM should consider including OGC in this plan if a hearing is requested on the power uprate. This plan should be issued about one week prior to the review completion date. The Plant PM should provide input to the Office of Public Affairs (OPA) for a press release about two weeks prior to the projected amendment review completion date. OPA typically issues the press release on the day the amendment review is completed (or shortly thereafter). The Plant PM is responsible for preparing an EDO Daily Note or an EDO Weekly Highlight upon completion of NRC’s review of a power uprate application or upon withdrawal of the amendment application by the licensee.

The Plant PM will ensure (via e-mail) that the Regional power uprate point-of-contact, at least one resident inspector at the plant with the power uprate, and the appropriate NRR Branch Chiefs have received the staff’s SE supporting the power uprate and are aware of any license conditions, regulatory commitments, and recommended areas for inspection sections in the SE, upon approval of a power uprate application. The Plant PM will inform the Lead PM when the SE has been communicated to the Regional and NRR individuals discussed above, with focus on the SE sections discussed above.

DORL management provides oversight to ensure that the Plant PMs are performing the duties discussed above and that the power uprate performance measures (i.e., review timeliness goals) discussed later in this office instruction are being met or there is adequate justification for not meeting them. DORL management ensures that DPR is kept informed on progress and issues regarding plant-specific power uprate applications.

<u>Technical Divisions/Branches</u>

The technical staff is responsible for conducting acceptance reviews per LIC-109, and for providing quality SE inputs and any recommended areas for inspection (typically for EPUs), on the agreed-upon schedule that was established with the Plant PM. If the technical staff identifies substantial technical issues beyond the scope of a typical power uprate request in the application, it should raise the issue immediately to management so that management can consider appropriate changes to the review schedule, including deviations from the standard power uprate review schedules shown in Appendix D. The technical staff will provide early notification to the Plant PM of any issue that may impact the review schedule (i.e., the SE input due date).

LIC-101 and RS-001 provide guidance on the outline/format of SE inputs. Examples of acceptable SE inputs are shown in Appendix B. These SE input examples were selected because they clearly described the changes, the regulatory requirements related to the changes, and explained why the staff's disposition of the changes satisfy regulatory requirements. In addition, these SE inputs were easy to read and certain portions of them reflect independent engineering judgements or analyses performed by the staff.

For complex technical issues, in order to obviate the need for multiple rounds of RAIs, the technical staff should consider audits (or working-level meetings) where they will enhance review efficiency. Previously, audits have been initially considered or actually held in the areas of reactor systems and nuclear performance reviews, flow-induced vibration reviews, chemical engineering reviews, and human performance reviews; but any area can be considered for an audit. Any technical information identified during the audit that is needed to support the staff's safety finding for the power uprate, needs to be formally submitted on the docket by the licensee.

The technical staff is responsible for providing briefings on power uprate technical issues to NRC management. The technical staff is responsible for providing timely inputs to the Plant PMs or the Lead PM to support their schedules for providing power uprate briefings or write-ups requested by NRC senior management.

Resource assumptions used for modeling power uprate reviews are shown in Appendix C. Individual applications may require more or less review time depending on the nature of the technical issues. Significant deviations from these estimates when performing power uprate reviews should be readily justified to NRC management upon request. The technical staff management (Branch Chief or higher) should periodically review the resource expenditures on power uprate reviews and propose any needed changes to these resource assumptions to PGCB. The changes should be based on historical resource expenditure data and future review expectations.

The technical branch and division management provide oversight to ensure that the technical staff is performing the duties discussed above and that the power uprate performance measures (i.e., review timeliness goals) discussed later in this office instruction are being met or there is adequate justification for not meeting them. Technical branch and division management ensure that quality SE inputs are provided to the Plant PMs and that they have consistent scope and depth of review, unless there is adequate justification to the contrary.

Technical branch or other division management determines whether all or a portion of the technical work should receive a peer review, in accordance with NRR Office Instruction ADM-405, "NRR Technical Work Product Quality and Consistency." ADM-405 provides criteria for technical work that should receive a peer review (e.g., issues that involve a new or first-of-kind review, are technically complex, or involve the use of new methodologies that could set new precedents).

Division of Inspection and Regional Support

The Reactor Inspection Branch is responsible for maintaining Inspection Procedure 71004,[3] Power Uprate, in consultation with the Regions. The Reactor Inspection Branch ensures that a power uprate point-of-contact(s) exists in each of the Regions.

NRR Management

NRR management shall resolve any disagreements between the Plant PMs, the Lead PM, and the technical staff regarding the scope, resources, and deadlines for power uprate safety reviews.

Regions

Inspection Procedure (IP) 71004 contains power uprate inspection requirements and guidance for the NRC Regional Offices. IP 71004 indicates that the NRC Regional Offices are responsible for developing an inspection plan and inspecting plants with approved power uprates greater than 7.5 percent above the CLTP limit, and that partial or complete implementation of IP 71004 should be considered for power uprates less than 7.5 percent above the CLTP limit. IP 71004 indicates that some inspection will take place before the power uprate is approved, while other inspection will take place afterwards.

IP 71004 requires that all planned team inspections that are selected to support completion of IP 71004 sample requirements, be annotated as such in the Reactor Program System. This designation will make inspectors and management aware of the link between the specific inspection and the associated power uprate.

Regarding documentation, IP 71004 requires power uprate inspection activities to be identified as such in inspection reports. Additionally, IP 71004 requires that a summary of power uprate inspections will be provided in an integrated inspection report once all required inspection samples are complete. The reason for these documentation requirements is so that power uprate related inspection activities can be easily identified.

6. PERFORMANCE MEASURES

The established performance timeliness goals are: 6 months for reviewing MUR power uprate applications, 9 months for reviewing SPU applications, and 12 months for reviewing EPU applications. These goals do not include the duration of the staff's acceptance review, which the staff conducts upon receipt of the initial application. Individual applications may require more or less review time depending on the nature of the technical issues. The staff will continue to ensure that the goal of protecting public health and safety is not compromised to meet these timeliness goals or resource assumptions in Appendix C.

[3] NRC Inspection Manual, Inspection Procedure 71004, “Power Uprate,” dated July 1, 2008 (ML081140192).

7. PRIMARY CONTACT

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8. RESPONSIBLE ORGANIZATION

NRR/DPR/PGCB

9. EFFECTIVE DATE

February 17, 2009

10. REFERENCES

RIS 2002-03, RIS 2007-24, RS-001, LIC-101, LIC-109, COM-109, COM-201, ADM-405

Enclosures:

1. Appendix A - Change History
2. Appendix B – Examples of SE Inputs
3. Appendix C – Resource Needs Assumptions
4. Appendix D – Power Uprate Milestones

APPENDIX A – CHANGE HISTORY

Office Instruction LIC-112

(Power Uprate Process)

LIC-112 Change History - Page 1 of 1			
Date	**Description of Changes**	**Method Used to Announce & Distribute**	**Training**
02/12/09	This is the initial issuance of Office Instruction (OI) LIC-112, "Power Uprate Process."	E-mail to NRR staff	Self-study

APPENDIX B

Office Instruction LIC-112 (Power Uprate Process)

Examples of SE Inputs

SE Input Example #1: The following excerpt is from NRC's SE on the Hope Creek EPU, dated May 14, 2008 (ML081230640, pages 9-12 of the SE). The definitions of the acronyms in the SE input below, if not set out below, are in the acronym section of the SE (i.e., see the acronym section in the referenced ML number shown above).

2.1.2 Pressure-Temperature Limits and Upper-Shelf Energy

Regulatory Evaluation

Pressure-temperature (P-T) limits are established to ensure the structural integrity of the ferritic components of the RCPB during any condition of normal operation, including anticipated operational occurrences (AOOs) and hydrostatic tests. The NRC staff's review of P-T limits covered the P-T limits methodology and the calculations for the number of EFPY specified for the proposed Hope Creek EPU, considering neutron embrittlement effects and using linear elastic fracture mechanics. The NRC's acceptance criteria for P-T limits are based on: (1) GDC-14, insofar as it requires that the RCPB be designed, fabricated, erected, and tested so as to have an extremely low probability of rapidly propagating fracture; (2) GDC-31, insofar as it requires that the RCPB be designed with margin sufficient to assure that, under specified conditions, it will behave in a non-brittle manner and the probability of a rapidly propagating fracture is minimized; (3) 10 CFR Part 50, Appendix G, which specifies fracture toughness requirements for ferritic components of the RCPB; and (4) 10 CFR 50.60, which requires compliance with the requirements of 10 CFR Part 50, Appendix G. Specific review criteria for the Hope Creek EPU are contained in SRP Section 5.3.2 and other guidance provided in Matrix 1 of Power Uprate Review Standard RS-001.[4]

Technical Evaluation

The ¼ T fluence is the fluence value at ¼ T from the Inside Diameter (ID) of the vessel with T being the vessel thickness. The ¼ T fluence is used for the evaluation of Pressure – Temperature (P – T) curves and Upper Shelf Energy (USE). The ¼ T fluence includes EPU conditions.

Upper-Shelf Energy (USE) Value Calculations

Appendix G of 10 CFR Part 50 provides the NRC's criteria for maintaining acceptable levels of USE for the reactor vessel beltline materials of operating reactors throughout the licensed lives of the facilities. The rule requires reactor vessel beltline materials to have a minimum USE value of 75 foot-pound force (ft-lb) in the unirradiated condition, and to maintain a minimum USE value above 50 ft-lb throughout the life of the facility, unless it can be demonstrated through analyses that lower values of USE would provide acceptable margins of safety against fracture equivalent to those required by Appendix G of Section XI to the ASME Code. The rule also mandates that the methods used to calculate USE values must account for the effects of neutron irradiation on the USE values for the materials and must incorporate any relevant reactor vessel surveillance capsule data that are reported through implementation of a plant's 10 CFR Part 50, Appendix H reactor vessel materials surveillance program.

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[4] ADAMS Accession No. ML033640024

The licensee for Hope Creek discussed the impact of the Hope Creek EPU on the Charpy USE values for the reactor vessel beltline materials in Section 3.2.1 of the PUSAR.[5] Table 3-2, "Hope Creek Upper Shelf Energy - 40 Year Life (32 EFPY)," pp 3-35 of the Hope Creek PUSAR, indicated that the projected Charpy USE for the limiting plate (intermediate shell plate, heat 5K3025) is 60 ft-lbs, and the projected Charpy USE for the limiting weld (intermediate-lower shell-to-intermediate shell circumferential submerged arc weld, heat D55733) is 60 ft-lbs. However, the NRC staff noted that in Table 3-2, heat 10024/1 for the low-pressure coolant injection (LPCI) nozzle forging specifies a copper content of 0.15 percent. In addition, the Hope Creek UFSAR, Appendix 5A, Tables 5A-5 and 5A-19 specifies a copper content of 0.14, while the NRC Reactor Vessel Integrity Database (RVID) specifies a copper content of 0.35 percent for the LPCI forging. In response to an RAI, the licensee, in its letter dated March 13, 2007,[6] confirmed that for heat 10024/1, the copper content is 0.14 percent. This is based on the General Electric Report GE-NE-523-A164-1294R1, Tables 7-2 and 7-3. The NRC staff confirmed that the copper content is 0.14 percent based on the report and will use the reported value to update the RVID copper value for this heat of material.

RG 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," has two methods for determining the percent reduction in Charpy USE. In Position 1.2, the percent reduction in Charpy USE is determined from Figure 2 in RG 1.99, Revision 2, which is based on the neutron fluence and the amount of copper in the material. In the second method, identified as Position 2.2, the percent reduction in Charpy USE is determined from surveillance data. RG 1.99, Revision 2 indicates surveillance data may be used for determining the Charpy USE when two or more credible surveillance data sets become available from the reactor. Since only one data set is presently available from the Hope Creek surveillance weld and surveillance plate, RG 1.99, Revision 2 would recommend that the Charpy USE be determined using Position 1.2. Using Figure 2 in RG 1.99, Revision 2, the staff determined that the percent reduction in Charpy USE based on an EOL neutron fluence of 5.3×10^{17} n/cm^2 (E > 1 MeV) was 11.1 percent for the plate material and the submerged arc weld material. Using the unirradiated values for the Charpy USE for the plate (75 ft-lbs) and the weld (68 ft-lbs) and the percent reduction determined using Figure 2 in RG 1.99, Revision 2, the Charpy USE at a neutron fluence of 5.3×10^{17} n/cm^2 (E > 1 MeV) is 66 ft-lb for the plate material and 60 ft-lb for the weld material. Since both the weld metal and plate material are projected to have Charpy USE greater than 50 ft-lb at EOL under Hope Creek EPU operating conditions, the reactor vessel materials satisfy the requirements of 10 CFR Part 50, Appendix G. As discussed in Section 2.1.1 of this SE, the surveillance data from Hope Creek (under the BWRVIP ISP) will be used to monitor the impact of neutron radiation on the Hope Creek beltline materials. In accordance with 10 CFR Part 50, Appendix G, the licensee is required to re-evaluate the impact of neutron radiation on Charpy USE when its surveillance data becomes available.

Pressure-Temperature Limit Calculations

Section IV.A.2 of 10 CFR Part 50, Appendix G requires that the P-T limits for operating reactors be at least as conservative as those that would be generated if the methods of calculation in the ASME Code, Section XI, Appendix G were used to calculate the P-T limits. The regulation also requires that the P-T limit calculations account for the effects of neutron irradiation on the P-T limit values for the reactor vessel beltline materials and incorporate any relevant reactor vessel

[5] Attachment 4, page 3-3 of PSEG Letter (LR-N06-0286) to NRC dated September 18, 2006, "Request for License Amendment Extended Power Uprate, Hope Creek Generating Station Facility, Operating License NPF-57, Docket No. 50-354" ADAMS Accession No. ML062680451
[6] PSEG Letter (LR-N-07-0035) to NRC dated March 13, 2007, "Response to Request for Additional Information - Request for License Amendment – Extended Power Uprate" ADAMS Accession No. ML070790508

surveillance capsule data that are required to be reported as part of the licensee's implementation of its 10 CFR Part 50, Appendix H reactor vessel materials surveillance program.

Section 3.2.1 of the PUSAR[7] indicates that the P-T limit curves contained in the technical specifications (TSs) remain bounding for Hope Creek EPU operating conditions and were approved in Hope Creek Amendment No. 157[8] dated November 1, 2004. Table 3-1 of the PUSAR (page 3-34), indicated that the adjusted reference temperature (ART) for the limiting material (intermediate shell plate, heat 5K3025) is 75 °F at a 1/4T fluence value of 3.7×10^{17} n/cm^2 (E > 1 MeV). This is consistent with the value referenced in the staff's November 1, 2004, safety evaluation which approved the P-T limit curves for 32 EFPY under Hope Creek EPU operating conditions. Therefore, the NRC staff agrees that the P-T limit curves contained in the TSs remain bounding for Hope Creek EPU operating conditions.

Conclusion

The NRC staff has reviewed the licensee's evaluation of the effects of the proposed Hope Creek EPU on the USE values for the reactor vessel beltline materials and P-T limits for the plant. The staff concludes that the licensee has adequately addressed changes in neutron fluence and their effects on the USE values for Hope Creek reactor vessel beltline materials and the P-T limits for the plant. The staff concludes that the Hope Creek beltline materials will continue to have acceptable USE values, as mandated by 10 CFR Part 50, Appendix G, through the expiration of the current operation license for the facility. The NRC staff further concludes that the licensee has demonstrated the validity of the current P-T limits for the proposed Hope Creek EPU operating conditions. Based on this, the NRC staff concludes that the proposed P-T limits will continue to meet the requirements of 10 CFR Part 50, Appendix G, and 10 CFR 50.60 and will enable the licensee to comply with GDC-14, and 31 following implementation of the proposed Hope Creek EPU. Therefore, the NRC staff finds the proposed Hope Creek EPU acceptable with respect to the TS P-T limits.

[7] Attachment 4, page 3-3 of PSEG Letter (LR-N06-0286) to NRC dated September 18, 2006, "Request for License Amendment Extended Power Uprate, Hope Creek Generating Station Facility, Operating License NPF-57, Docket No. 50-354" ADAMS Accession No. ML062680451
[8] ADAMS Accession No. ML042050079

SE Input Example #2: The following excerpt is from NRC's SE on the Susquehanna 1&2 EPU, dated January 30, 2008 (ML081000255, pages 100-107 of the SE). The definitions of the acronyms in the SE input below, if not set out below, are in the acronym section of the SE (i.e., see the acronym section in the referenced ML number shown above).

2.6.1 Primary Containment Functional Design

Regulatory Evaluation

The containment encloses the reactor system and is the final barrier against the release of significant amounts of radioactive fission products in the event of an accident. The NRC staff's review of the primary containment functional design covered (1) the temperature and pressure conditions in the drywell and wetwell that would result from a spectrum of postulated LOCAs, (2) the differential pressure across the operating deck for a spectrum of LOCAs (Mark II containments only), (3) suppression pool dynamic effects during a LOCA or following the actuation of one or more RCS SRVs, (4) the consequences of a LOCA occurring within the containment (wetwell), (5) the capability of the containment to withstand the effects of steam bypassing the suppression pool, (6) the suppression pool temperature limit during RCS SRV operation, and (7) the analytical models used for containment analysis. The NRC's acceptance criteria for the primary containment functional design are based on (1) GDC 4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents and that such SSCs be protected against dynamic effects, (2) GDC 16, "Containment Design," insofar as it requires that reactor containment be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment, (3) GDC 50, "Containment Design Basis," insofar as it requires that the containment and its associated heat removal systems be designed so that the containment structure can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated temperature and pressure conditions resulting from any LOCA, (4) GDC 13, "Instrumentation and Control," insofar as it requires that instrumentation be provided to monitor variables and systems over their anticipated ranges for normal operation and for accident conditions, as appropriate, to assure adequate safety, and (5) GDC 64, "Monitoring Radioactivity Releases," insofar as it requires that means be provided to monitor the reactor containment atmosphere for radioactivity that may be released from normal operations and from postulated accidents. SRP Section 6.2.1.1.C contains specific review criteria.

Technical Evaluation

The primary containments for both SSES Unit 1 and Unit 2, as described in Section 3.8 of the SSES Unit 1 and 2 FSAR (Revision 58), form an enclosure for the RV, the reactor coolant recirculation loops, and other branch connections of the RCS. The major elements of the primary containment are the drywell, the pressure suppression chamber that stores a large volume of water, the drywell floor that separates the drywell and the suppression chamber, the connecting vent pipe system between the drywell and the suppression chamber, isolation valves, the vacuum relief system, and the containment cooling systems and other service equipment.

The primary containment is in the form of a truncated cone over a cylinder section, with the drywell in the upper conical section and the suppression chamber in the lower cylindrical section. The primary containment is made of reinforced concrete lined with welded steel plate. A steel domed head is provided for closure at the top of the drywell.

The proposal to operate at EPU conditions requires that safety analyses for those DBAs whose results depend on power level be recalculated at the higher power level. The containment design basis is primarily established based on the LOCA and the actuation of the RV SRVs and their discharge into the suppression pool.

The SSES Unit 1 and 2 FSAR reports the results of short-term and long-term containment analyses. The short-term analysis is directed primarily at determining the drywell pressure response during the initial blowdown of the RV inventory to the containment following a large break of a recirculation line inside the drywell. The long-term analysis is directed primarily at the suppression pool temperature response, considering the decay heat addition to the suppression pool. The effect of power on the events yielding the limiting containment pressure and temperature responses is described below.

The reevaluation of the long-term containment LOCA response reflects two changes to the SSES Unit 1 and 2 licensing basis. These changes are (1) crediting the presence of passive heat sinks and (2) the use of the ANSI/ANS 5.1-1979 decay heat model, which has a 2-sigma (σ) uncertainty instead of the ANS 5 model which has a 20-percent/10-percent uncertainty. Both of these changes are consistent with GE containment analyses accepted by the NRC for other BWR licensing actions. Both changes are acceptable for SSES Units 1 and 2 as discussed below.

Short-Term LOCA Analysis

The short-term analysis covers the blowdown period during which the maximum drywell pressure, maximum wetwell pressure, and maximum differential pressure between the drywell and the wetwell occur. The short-term LOCA analysis is performed for the limiting DBA LOCA, which assumes a double-ended guillotine break of a recirculation suction line, to show that the peak drywell pressure and temperature remain below the drywell design pressure of 53 psig and the drywell design temperature of 340 °F. The short-term analysis covers the blowdown period during which the maximum drywell pressure and maximum differential pressure between the drywell and suppression chamber occur. These analyses were performed at 2 percent above the EPU-rated thermal power (RTP), using analytic methods approved for EPUs. The RV steam dome pressure remains constant at its pre-EPU value. The EPU is therefore a CPPU. The licensee used the LAMB computer code (Reference 46) for the short-term mass and energy release and the M3CPT computer code (Reference 59) for the containment response. The power uprate methods approved by the NRC permit the use of either the M3CPT computer code or the LAMB computer code to calculate the mass and energy release from the postulated pipe break into the drywell (Reference 10).

The short-term containment analyses make several conservative assumptions. The reactor is assumed to be operating at 2 percent above the RTP to include instrument uncertainty effects, consistent with RG 1.49, “Power Levels of Nuclear Power Plants.” The suppression pool level and mass are at values corresponding to the maximum TS limit. The recirculation suction line is assumed to instantaneously undergo a double-guillotine break. The vessel depressurization flow rates are calculated using the Moody critical flow model (Reference 60) which maximizes the mass flow into the drywell. The MSIV closure time is minimized so as to maintain RV pressure which in turn maximizes the break flow into the drywell. The fluid flowing through the drywell-to-wetwell vents is assumed to be a homogenous mixture of the fluid in the drywell. Thus, the flow contains liquid droplets. The presence of these liquid droplets increases the pressure drop of the flow through the vents and therefore increases the drywell pressure. The

FSAR analyses assume that there is no heat loss from the gases inside the primary containment. In reality, condensation of steam on the drywell surfaces would be expected. Neglecting this heat transfer is conservative for the short-term analyses.

The licensee has revised the assumed behavior of the FW flow into the vessel following the recirculation line break. The current licensing basis assumes that FW flow into the vessel continues at a flow rate which decreases with time (see FSAR Figure 6.2-9a). The CPPU analysis assumes reactor FW flow into the vessel remains at full rated flow for 10 seconds. The licensee has demonstrated that this assumption is more conservative than the current licensing basis (Reference 61) and it is, therefore, acceptable.

The licensee also made changes that reduce conservatism. The method of inputting break flow data into the M3CPT code has been revised. The licensee stated that the mass flow rate is still conservative and that a certain amount of overconservatism has been removed. Since the break flow rate remains conservative, the NRC staff finds this change acceptable.

Table 4-1 of the PUSAR (Reference 1) presents the results of these analyses at EPU and the acceptance criteria. The short-term portion of this table is reproduced below.

SSES Unit 1 and 2 Short-Term LOCA Containment Performance Results

Parameter	Current Licensed Thermal Power from FSAR	Using CPPU Analysis Method with CLTP Assumptions	CPPU	Design Limit
Peak Drywell Pressure (psig)	44.6	47.9	48.6	53
Peak Drywell Air Space Temperature (°F)	320*	337*	337*	340
Peak Drywell-to-Wetwell (Down) Differential Pressure (psid)	27.0	25.9	25.6	28

* These peak drywell temperatures are for a large, double-ended guillotine break of a main steamline.

The table allows separation of the effects on important containment parameters that result from the power uprate and those that result from the change in analysis assumptions. The licensee's June 4, 2007, response to NRC RAI 3, describes the reasons for the differences between the parameters listed in this table. The differences in the short-term analyses shown in this table are primarily the result of different assumptions in the initial drywell and suppression chamber pressures.

The licensee stated that the decrease in peak differential pressure is primarily the result of a GE proprietary change in the method for calculating the wetwell pressures associated with the pool swell phenomenon. The NRC staff finds this change to be acceptable.

P_a is the pressure at which containment leakage rate testing is performed. It is defined in Appendix J to 10 CFR Part 50, as the calculated peak containment internal pressure related to the design-basis LOCA. The licensee proposed to revise P_a in SSES Unit 1 and 2 TS 5.5.1.2, Primary Containment Leakage Rate Testing Program, to 48.6 psig. The NRC staff finds this acceptable since P_a, the calculated peak containment internal pressure related to the design-basis LOCA for the EPU, is determined with acceptable methods and assumptions.

The licensee also proposed to change TS 3.6.1.3.12, which requires leakage rate testing of the MSIVs, to revise the test pressure from 22.5 psig (which is half of the current value of P_a) to 24.6 psig (which is half of the proposed value of P_a). Since the value of P_a is acceptable, this change is acceptable.

Based on the use of acceptable calculation methods and conservative assumptions and results less than the design containment pressure and temperature, the NRC staff finds the SSES Unit 1 and 2 short-term containment response at EPU to be acceptable.

<u>Long-Term LOCA Analysis</u>

The long-term LOCA analysis was performed for the DBA LOCA at 2 percent above the EPU RTP. The SHEX computer code (Reference 62) is used for the analysis of the peak suppression pool temperature, long-term peak wetwell pressure, and peak wetwell air temperature. The NRC has accepted this computer code for previous power uprate applications.

After 600 seconds into the accident, it is assumed that the operator actuates the RHR heat exchangers using the RHRSWS as the heat sink. The initial suppression pool level is at its minimum value. The calculation includes the effects of decay heat, stored energy, and energy from the metal water reaction.

The licensee previously used the ANS 5-1971 decay heat model with a +20 percent/10 percent margin for uncertainty (Reference 61). For the EPU, the licensee proposes to use the ANSI/ANS 5.1-1979 decay heat model with a 2-sigma uncertainty added (Reference 62). The licensee incorporated the guidance of GE Service Information Letter (SIL) 636, Revision 1 (Reference 63), which recommends accounting for additional actinides and activation products, which further increases the predicted decay heat. Because the NRC staff has accepted the ANSI/ANS 5.1-1979 decay heat model with a two-sigma uncertainty in previous EPU reviews, as well as other safety analyses, it is acceptable for SSES Units 1 and 2.

The licensee currently credits the suppression pool as the only passive heat sink available in the containment system. For the EPU, the licensee proposes to credit heat transfer from the containment atmosphere to passive heat sinks in the drywell, suppression chamber air space, and suppression pool. The NRC staff has reviewed the licensee’s approach and finds it conservative and acceptable.

The RHR system heat exchanger removes heat from the suppression pool. When the energy removal rate of the RHR system exceeds the energy addition rate from the decay heat and pump heat, the containment pressure and temperature reach a second peak value and decrease gradually.

An important parameter characterizing the performance of the suppression pool is the K value of the RHR heat exchanger. For SSES Units 1 and 2, K equals 317.5 British thermal units per second-degrees Fahrenheit (Btu/s-°F). This is the value assumed in the current licensing-basis analysis for containment response. The RHR heat exchangers are periodically tested according to the recommendations of NRC GL 89-13 (Reference 65). This testing ensures that the heat exchangers meet or exceed this K value.

The long-term LOCA analysis demonstrates that the peak suppression pool temperature and wetwell pressure remain below their respective design limits. Table 4 -1 of the PUSAR presents the results of these analyses and the acceptance criteria. The relevant portions of this table are reproduced below.

Susquehanna Long-Term Containment Performance Results
(At Extended Power Uprate)

Parameter	CLTP from FSAR	Using CPPU analysis method with CLTP assumptions	CPPU	Design Limit
Peak Bulk Pool Temperature (°F)	203	192	211.2	220
Peak Wetwell Pressure (psig)	35.3	36.7	36.5	53

The wetwell pressure peaks early in the event and then peaks again around the time at which the wetwell temperature peaks. This table presents the value of the second (lower) peak pressure.

The EPU peak suppression pool temperature of 211.2 °F is less than the suppression pool design temperature of 220 °F. Since the licensee used acceptable calculation methods and conservative assumptions and the calculated values are below the design limits, the long-term containment calculations for extended power conditions are acceptable.

Hydrodynamic Loads

Part of the containment design basis is the acceptable response of the containment to hydrodynamic loads associated with the discharge of reactor steam and drywell nitrogen into the suppression pool following a LOCA or the discharge of reactor steam following actuation of the SRVs. The licensee used analytical and empirical methods developed by the ad hoc Mark II Owners' Group and approved by the NRC staff in NUREG-0808 (Reference 66) to address these issues for SSES Units 1 and 2.

The licensee must ensure, as part of the power uprate evaluation, that these analyses remain bounding for operation at CPPU conditions. This is done for the LOCA by means of short-term calculations of the pressure and temperature response to a double-ended break of an RCS recirculation line. The key parameters are the drywell and wetwell pressure, vent flow rates, and the suppression pool temperature.

The licensee considered LOCA-induced loads such as the submerged boundary loads during vent clearing, pool swell loads, and LOCA steam condensation pool boundary loads (CO and chugging). Vent clearing refers to the ejection of water in the downcomers caused by drywell pressurization as a result of the LOCA. Vent clearing produces pressure loads on the containment basemat and the submerged suppression chamber walls. The NRC acceptance criteria stipulate an overpressure criterion on the basemat and walls below the vent exit of 24 psi. The licensee stated that an evaluation of the specified load concludes that the 24 psi overpressure is not exceeded.

The pool swell loads are a function of the initial drywell pressurization rate during a LOCA. The licensee stated that the results of the CPPU pool swell analysis are bounded by the current analysis. The licensee discussed the reasons for this in response to an NRC RAI (Reference 61). The NRC staff finds the licensee's explanation acceptable, since it is based on the use of the NRC-approved computer code (currently designated as PICSM) and the assumptions are consistent with the NRC recommendations of NUREG-0808 and NUREG-0487 (Reference 67). These reports reviewed the Mark II containment hydrodynamic loads testing and analyses and provided acceptance criteria acceptable to the NRC staff for plant-specific analyses.

Condensation loads increase with higher suppression pool temperature and/or a higher vent mass flow rate. The licensee compared the break flow rate (and hence the vent flow) for CPPU conditions with the vent flow calculated for the GKM-II-M test. (GKM II was a full-scale, single-vent test facility used by the licensee to obtain CO and chugging data.) The CO loads remain bounding. Therefore, the CO loads for the CPPU are acceptable.

The licensee's evaluation of containment hydrodynamic loads as a result of a LOCA is in accordance with the EPU topical report (Reference 10) and shows acceptable results. These results are therefore conservative and acceptable for the EPU.

<u>Safety/Relief Valve Loads</u>

The dynamic loads on the suppression pool due to the discharge of steam from SRVs are part of the containment design basis. The SRV loads evaluated for the CPPU are loads on the quenchers, quencher supports, and SRV discharge lines; loads on the submerged boundary of the suppression pool; and loads on submerged structures in the suppression pool.

The parameters that affect the SRV loads, the RV pressure, the SRV opening and closing setpoints, the submergence of the quenchers, the line air volume, and the automatic depressurization system (ADS) setpoints do not change for the CPPU. Therefore, the CPPU does not affect the SRV loads.

<u>Local Pool Temperature with MSRV Discharge</u>

NUREG-0783 (Reference 68) specifies a local pool temperature limit for SRV discharge because of concerns resulting from unstable condensation observed at high pool temperatures in BWRs without quenchers. The licensee indicated that an evaluation of the SSES Unit 1 and 2 peak local suppression pool temperature for EPU shows that the temperature meets the NUREG-0783 criteria. The SRV flow capacities and the configuration of the SSES Unit 1 and 2 T-quenchers remain unchanged for EPU, and the predicted local pool temperatures remain below the NUREG-0783 limit. Therefore, the SSES Unit 1 and 2 peak local suppression pool temperature is acceptable for the EPU conditions.

The licensee has not proposed any changes to instrumentation and controls provided to monitor and maintain variables within prescribed operating ranges. The licensee also has not proposed any changes to instrumentation used to monitor the reactor containment atmosphere for radioactivity that may be released from normal operations and from postulated accidents.

Conclusion

The NRC staff has reviewed the licensee's assessment of the containment temperature and pressure transient and concludes that the licensee has adequately accounted for the increase of mass and energy resulting from the proposed EPU. The NRC staff further concludes that containment systems will continue to provide sufficient pressure and temperature mitigation capability to ensure that containment integrity is maintained. The NRC staff also concludes that containment systems and instrumentation will continue to be adequate for monitoring containment parameters and release of radioactivity during normal and accident conditions and the containment and associated systems will continue to meet the requirements of GDC 4, 13, 16, 50, and 64 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to primary containment functional design.

SE Input Example #3: The following excerpt is from NRC's SE on the Beaver Valley 1&2 EPU, dated July 19, 2006 (ML061720376, pages 96-99 of the SE). The definitions of the acronyms in the SE input below, if not set out below, are in the acronym section of the SE (i.e., see the acronym section in the referenced ML number shown above).

2.8.1 Fuel System Design (EPULR Sections 4.3, and 6.0)

Regulatory Evaluation

The fuel system consists of arrays of fuel rods, burnable poison rods, spacer grids and springs, top and bottom nozzles, and reactivity control rods. The NRC staff reviewed the fuel system to ensure that (1) the fuel system is not damaged as a result of normal operation and anticipated operational occurrences (AOOs), (2) fuel system damage is never so severe as to prevent control rod insertion when it is required, (3) the number of fuel rod failures is not underestimated for postulated accidents, and (4) coolability is always maintained. The staff's review covered fuel system damage mechanisms, limiting values for important parameters, and performance of the fuel system during normal operation, AOOs, and postulated accidents. The NRC's acceptance criteria are based on (1) 10 CFR 50.46, insofar as it establishes standards for the calculation of ECCS performance and acceptance criteria for that calculated performance; (2) GDC 10, insofar as it requires that the reactor core be designed with appropriate margins to assure that specified acceptable fuel design limits (SAFDLs) are not exceeded during any condition of normal operation, including the effects of AOOs; (3) GDC 27, insofar as it requires that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the ECCS, of reliably controlling reactivity changes under postulated accident conditions, with appropriate margins for stuck rods, to assure the capability to cool the core is maintained; and (4) GDC 35, insofar as it requires that a system to provide abundant emergency core cooling be provided to transfer heat from the reactor core following any LOCA. Specific review criteria are contained in SRP Section 4.2 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

To support the EPU, the fuel assembly design was changed from the Vantage 5H (V5H) design to the Robust Fuel Assembly (RFA) design. The RFA fuel geometry/characteristics remain the same as the V5H fuel assemblies. The major change to the fuel assembly from V5H to RFA is the redesigned mid-grids, the addition of intermediate flow mixing grids, and thicker instrument and guide tubes. The BVPS cores have been completely transitioned from V5H to RFA fuel assemblies. The licensee states that previously burned V5H fuel assemblies may be reinserted as part of a cycle-specific reload pattern. The V5H fuel design is mechanically and hydraulically compatible with the RFA fuel design.

Structurally, the V5H fuel assembly design is very similar to the VANTAGE+ fuel assembly design [28]. The most significant difference is the implementation of a new cladding material, ZIRLO™. BVPS-1 and 2 received license amendments permitting the use of VANTAGE+ fuel on May 23, 1997 [29] and September 13, 1996 [30], respectively.

The RFA/RFA-2 fuel designs are modifications of the physical structure of the 17x17 VANTAGE+ fuel assembly design. The RFA/RFA-2 modifications were licensed under the Westinghouse fuel criteria evaluation process (FCEP) [31]. The FCEP is an NRC-approved process whereby Westinghouse may make minor changes to its fuel designs without prior NRC approval. Westinghouse is required to notify the NRC when such changes are made. FCEP notifications for the RFA and RFA-2 fuel designs were made to the NRC on September 30, 1998 [32] and August 31, 2001 [33], respectively. As with any other change, the licensee must then evaluate the change and implement it either by using the 10 CFR 50.59 change process or by requesting a license amendment.

Since the RFA and RFA-2 fuel systems at BVPS-1 and 2 have already been evaluated for use at the currently licensed RTP, this review will focus on the effects of the EPU.

The EPU will cause the fuel operating temperatures and the fuel assembly average burnup to increase. In addition, the best-estimate flow will increase due to (1) the RSGs for BVPS-1, and (2) the change in SG tube plugging limits for BVPS-1 and 2. Therefore, the fuel system design criteria that must be evaluated are: stress and strain, fatigue, grid-to-rod fretting, corrosion, dimensional changes, rod internal pressure, fuel assembly lift forces, and vibration.

Fuel System Damage

The licensee evaluated the EPU for its affect on fuel system damage due to clad stress and strain, corrosion, assembly grid-to-rod fretting, internal rod pressure, and hydraulic loads. The licensee used an NRC-approved fuel performance model [34]; [35]; [36] to evaluate the impact of the EPU on these criteria. The licensee's analysis shows that the EPU core will not impact the fuel's capability to meet clad stress and strain limits, and fatigue limits for the EPU conditions. The licensee's analysis also shows that the EPU's increased operating temperatures for the clad, due to the increased rod average power rating, will not impact the fuel's capability to meet corrosion limits for both the ZIRLO™ and Zircaloy-4 clad fuel. The licensee determined that the propensity for crud deposition and chemical plate-out on the cladding, with proper chemistry control, will not significantly increase under EPU conditions, and that the internal rod pressure acceptance criterion (no increase in the diametrical gap due to clad creep during steady-state operation or for DNB propagation to occur) is satisfied. Finally, the licensee determined that fuel assembly hold down spring capacity is still acceptable, given the increased up-lift force associated with the best-estimate RCS flow and the increased fuel assembly growth due to the higher assembly average burnup. Based on the results of the licensee's analysis using the NRC-approved fuel performance model which demonstrates that the EPU core will not result in fuel damage, the NRC staff finds the licensee's fuel damage assessment acceptable with respect to EPU.

Fuel Rod Failure

Internal hydriding and cladding collapse are primarily a result of deficiencies in the manufacturing process, which is not an EPU-related factor, and therefore, not considered further in this review.

Test results from the vibration investigation and pressure drop experimental research (VIPER) loop for the RFA/RFA-2 fuel designs continue to bound the BVPS-1 and 2 assemblies operating under EPU conditions. The transient analyses submitted in the EPULR demonstrate that the SAFDLs are not exceeded for normal operation and AOOs, and that the number of predicted fuel rod failures is not underestimated for postulated accidents.

Fuel Coolability

The licensee evaluated the EPU for its affect on fuel system embrittlement and fuel rod ballooning. The licensee used an NRC-approved fuel performance model [34]; [35]; [36] to evaluate the impact of the EPU on these criteria. The licensee's analysis shows that the hydrogen pickup level in the cladding will be less than the acceptance limit. The licensee determined the internal rod pressure acceptance criterion to prevent DNB propagation is met, thereby preventing fuel rod ballooning. The transient analyses submitted in the EPULR demonstrate that the fuel system damage is never so severe as to prevent control rod insertion when it is required, that the number of predicted fuel rod failures is not underestimated for postulated accidents, and that coolability is always maintained. Based on the licensee's analysis using an NRC-approved fuel performance model which demonstrates that fuel rod ballooning is not expected to occur and control rod insertion will not be affected, the NRC staff finds the licensee's assessment of fuel coolability to be acceptable.

Conclusion

The NRC staff has reviewed the licensee's analyses related to the effects of the proposed EPU on the fuel system design of the fuel assemblies, control systems, and reactor core. The staff concludes that the licensee has adequately accounted for the effects of the proposed EPU on the fuel system and demonstrated that (1) the fuel system will not be damaged as a result of normal operation and AOOs, (2) the fuel system damage will never be so severe as to prevent control rod insertion when it is required, (3) the number of fuel rod failures will not be underestimated for postulated accidents, and (4) coolability will always be maintained. Based on this, the staff concludes that the fuel system and associated analyses will continue to meet the requirements of 10 CFR 50.46, GDCs 10, 27, and 35 following implementation of the proposed EPU. Therefore, the staff finds the proposed EPU acceptable with respect to the fuel system design.

APPENDIX C - RESOURCE NEEDS ASSUMPTIONS USED IN THE MODELS FOR POWER UPRATES[1]
(in hours)

	MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATE	STRETCH POWER UPRATE	EXTENDED POWER UPRATE
DORL	260	330	580
EICB - Instrumentation & Controls	200	80	160
EEEB - Electrical Engineering	40	80	260
CVIB - Vessel & Internals Integrity	40	100	170
CPNB - Piping & NDE	40	40	100
CSGB - SG Tube Integrity & Chemical Engineering	40	100	170
EMCB & CPTB - Mech. & Civil Eng., Component Perf. & Test	80	160	360
SBPB - Balance of Plant	5	120	390
AFPB - Fire Protection	5	80	160
APLA - PRA Licensing (Risk Evaluation)	0	0	400
SCVB & AADB - Containment & Ventilation, & Accident Dose	45	280	600
SNPB & SRXB - Nuclear Perf. & Code, & Reactor Systems	200	400	1000
EQVB - Quality & Vendor	0	0	240
IRIB - Health Physics	0	0	80
IOLB - Operator Licensing & Human Performance	5	10	190
RERB - Environmental	0	20	140
ITSB - Technical Specifications	40	40	40
TOTAL	**1000**	**1840**	**5040**

Note 1: This table is for reference only. Official values are found in the Operating Level Report. Change to this table does not constitute change to this office instruction.

APPENDIX D – POWER UPRATE MILESTONES[1]

POWER UPRATE MILESTONES	approximate - from application date (except for last 2 lines)	approximate - from application date (except for last 2 lines)	approximate – from application date (except for last 2 lines)
	MUR	**SPU**	**EPU**
REQUESTED MILESTONES			
Acceptance Review to PM	3 weeks from receipt[2]	3 weeks from receipt[2]	3 weeks from receipt[2]
RAI/draft SE to PM	2 months	3.5 months	4.5 months
SE Input to PM	4 months	7 months	8 months
Prepare for ACRS Sub-Com.	N/A	7.5 months (if needed)	10 months
Prepare for ACRS Full Com.	N/A	7.5 months (if needed)	10 months
MANAGER's MILESTONES			
Acceptance Review to Licensee	4 weeks from receipt[2]	4 weeks from receipt[2]	4 weeks from receipt[2]
Initial Notice to Fed Register	2 months	2 months	2 months
RAI Issued to Licensee	2.5 months	4 months	5 months
RAI Response from Licensee	3.5 months	5.5 months	6.5 months
Issue Draft EA	4 months, if needed	7 months, if needed	10 months
Issue Final EA	5.5 months if needed	8.5 months if needed	11.5 months
Prepare Draft SE/Send to ACRS	N/A	7.5 months (1 month before ACRS subcommittee)	9.5 months (1 month before ACRS subcommittee)
Issue Proprietary Determination Letter	2 months from incoming, rolling as needed	2 months from incoming, rolling as needed	2 months from incoming, rolling as needed
Issue License Amendment	6 months*	9 months*	12 months*
Issue Press Release	6 months*	9 months*	12 months*
***from NRC acceptance**			

Notes: 1. This table is for reference only. Change to this table does not constitute change to this office instruction.
2. Receipt of application is defined as when it is available in the Agencywide Documents Access and Management System.