February 18, 1997

FOR: The Commissioners

FROM: Hugh L. Thompson, Jr. /s/
Acting Executive Director for Operations

SUBJECT: RESPONSE TO OIG EVENT INQUIRY 96-04S REGARDING MAINE YANKEE

PURPOSE:

To inform the Commission of the staff's actions to address the findings of the report of the Office of the Inspector General (OIG) entitled "The NRC Staff's Actions Related to Regulation at Maine Yankee."

BACKGROUND:

By memorandum dated May 8, 1996, the Inspector General submitted to the Commission the results of its event inquiry concerning the NRC staff's actions associated with small break loss-of-coolant accident (SBLOCA) analyses of the emergency core cooling systems (ECCS) to support two amendments increasing the rated thermal power at which Maine Yankee Atomic Power Station (MYAPS or Maine Yankee) could operate. The OIG inquiry was initiated as a result of an allegation in December 1995 regarding certain SBLOCA analyses for MYAPS performed by Yankee Atomic Electric Company (YAEC) for Maine Yankee Atomic Power Company (MYAPCo). In SECY-96-135, "Response to Event Inquiry - Maine Yankee Atomic Power Station (Case No. 96-04S)," dated June 21, 1996, the staff informed the Commission of its plans to address the findings of the May 8, 1996, OIG report entitled "The NRC Staff's Actions Related to Regulation at Maine Yankee." In SECY-96-135, the staff committed to report the results of its reviews and the status of the staff's corrective actions to the Commission.

DISCUSSION:

Recognizing the potential for safety concerns similar to those regarding SBLOCA analyses and power uprates at Maine Yankee arising at other power reactors, the staff started in early 1996 to address many of the issues subsequently raised in the OIG report. A team was formed within the Office of Nuclear Reactor Regulation (NRR) to determine the lessons learned from the problems identified with the ECCS analysis for MYAPS and to provide recommendations for improving the staff's processes. The team's plan was attached to SECY-96-135. This team was directed to address (1) the technical review processes for computer code applications and for power uprate amendment requests; (2) coordination between technical and projects staffs, including technical staff concurrence; (3) delegation of signature authority; (4) tracking and verification of licensee commitments; and (5) closeout of selected Three Mile Island (TMI) Action Plan items at other power reactors for potential vulnerabilities similar to those identified in the Maine Yankee case. On May 8, the OIG issued its report. Team activities continued and the team's plan formed the basis for SECY-96-135.

The OIG report documented nine observations. In SECY-96-135, the staff specifically addressed OIG Observation No. 4, i.e., that NRC staff on distribution for NRC correspondence had the opportunity to question that correspondence. The team's report, dated December 1996, (Attachment 1) addresses the issues raised in the remaining observations. Some of the team's recommendations regarding signature authority and handling of unsolicited information have already been addressed in revisions to applicable NRR Office Letters. The team's report is under consideration by NRR management and plans to address the remaining recommendations are being developed.

At the direction of Chairman Jackson, the NRC staff performed an Independent Safety Assessment (ISA) of MYAPCo during July and August of 1996. Chairman Jackson issued a report documenting the ISA team's findings on October 7, 1996. By memorandum dated November 27, 1996, the Executive Director for Operations (EDO) directed the staff to develop plans to resolve generic and plant-specific issues raised in the ISA report. These issues include the adequacy of the staff's analytic code review and power uprate review processes. The recommendations of the Maine Yankee lessons learned team in these areas will be considered in the development of NRR's plans to resolve the related ISA issues. NRR will develop a written summary of the plan and schedule for the eight ISA issues for which NRR has been assigned lead responsibility. The staff's plans and schedules to address all of the recommendations in Attachment 1 will also be included in the summary schedule. The summary schedule will consider the resources needed to appropriately address the team's recommendations consistent with their priority in relation to other ongoing staff activities. As requested in the EDO's memorandum of November 27, 1996, the summary should be provided to the EDO in late February 1997. This summary will be provided to the Commission after review by the EDO and coordination of the resource implication with the CFO.

Coordination Statement

The Office of the General Counsel has reviewed this paper and has no legal objection.

SUMMARY:

The Maine Yankee lessons learned team reviewed the staff's processes pertaining to the review of code applications and power uprates, closeout of TMI Action Plan items, and the role and responsibilities of the NRR project manager. These issues were also raised in the OIG report. No immediate safety concerns were identified. Several recommendations were made for improving the staff's processes. Some of the recommendations have already been incorporated into NRR Office Letters. Details of the implementation of the remaining recommendations are being developed. The team's report is currently under management review to determine the appropriate scope, resources and priority for following up on these recommendations. A summary of the staff's plan should be provided to the Commission in late March 1997.

Hugh L. Thompson, Jr.
Acting Executive Director for Operations

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Attachment: 1. Report of the Maine Yankee Lessons Learned Task Group

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APPENDIX

PLAN TO ADDRESS LESSONS LEARNED FROM MAINE YANKEE EXPERIENCE

ATTACHMENT

REPORT OF THE
MAINE YANKEE LESSONS LEARNED
TASK GROUP

Issued: December 1996

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Table of Contents

EXECUTIVE SUMMARY

In December 1995, the staff received allegations that the licensee for Maine Yankee Atomic Power Station had knowingly performed inadequate analyses of the emergency core cooling system (ECCS) and the containment in support of two power uprate amendments. The fact that the conditions asserted in these allegations were not identified by the NRC staff implies possible significant weaknesses in the NRC review and approval processes used in the subject power uprates. (These allegations also implied significant deficiencies in the licensee's performance that the NRC staff is addressing separately from this activity.)

As a result of the allegations and a subsequent assessment by the NRC staff, the Office of Nuclear Reactor Regulation's (NRR) Associate Director for Technical Review (ADT), in a memorandum dated January 5, 1996, requested that the Director of NRR's Division of Systems Safety and Analysis (DSSA) address issues arising from the Maine Yankee experience. In response to ADT's request, the Maine Yankee Lessons Learned Task Group (the team) was formed, and in a memorandum dated April 25, 1996, the Director of DSSA provided a plan for addressing the issues. This report presents the results of that effort.

Review and Followup of Analytical Codes and Methodologies

The NRC staff determined that plant-specific application of RELAP5YA for Maine Yankee did not fully conform to the requirements of 10 CFR 50.46 and 10 CFR 50 Appendix K, nor had it been applied in a manner conforming to the conditions specified in the staff's SER, as necessary for NRC acceptance of the use of RELAP5YA for small-break loss-of-coolant accident (SBLOCA) analyses for Maine Yankee. As a result, the staff concluded that the SBLOCA portion of the ECCS analyses performed for Maine Yankee Cycle 15 did not conform to the requirements of 10 CFR 50.46.

The team performed an assessment of the code review process, the code modification process, and staff followup of vendor and licensee code implementation. Options for a catalogue of codes were evaluated. The assessment indicated a need for improved consistency and uniformity in the submittal and more formal staff guidance in the review of code applications. In addition, staff verification of licensee and vendor implementation should be improved.

Power Uprate Review Process

The allegations on Maine Yankee included the assertion that inadequate analyses were performed by YAEC to support two power uprates (license amendments to increase the plant's licensed maximum reactor power level) for Maine Yankee. This assertion implies that the staff's reviews of the Maine Yankee power uprate applications were inadequate. To assess the adequacy of the staff's power uprate review process, the team examined past power uprate reviews and assessed them against current practice.

The wide variation in the scope and depth of the completed power uprate reviews shows the need for standardization of license amendment application reviews and integration of technical review conclusions into licensing and licensing basis documents. In addition, although the team did not identify any immediate safety concerns, the uprate reviews that have been conducted warrant a revisit to confirm the general conclusions of the reviews.

Closeout of TMI Action Items

A document search was conducted to locate and compile the closeout documentation for TMI Action Plan Items II.K.3.5, II.K.3.30, and II.K.3.31 for all plants currently holding an operating license. Documents were examined to develop a general understanding of the review process applied to each item by the staff and to identify potential problem areas in the closeout. Specifically, technical staff involvement in the closeout of each item for each affected plant was assessed, along with management oversight of the closeout documents and safety evaluations in the closeouts. Further inspection of licensee and vendor implementation of SBLOCA evaluation models is recommended. Also, for several facilities, the team recommends staff verification of the

applicability of the generic ECCS evaluation model.

Staff Interfaces

The team assessed various staff guidance documents, including Management Directives, NRR Office Letters, and the Project Manager's Handbook (NUREG/BR-0073, Revision 1), for guidance regarding documentation of communications with licensees, technical staff involvement, signature authority, and tracking of licensee commitments. In addition to evaluating the adequacy of staff guidance documents, current staff performance was benchmarked by reviewing selected documents generated by the projects organization within NRR during the first half of calendar year (CY) 1996. The documents reviewed included a broad sampling of licensing action approvals and denials, and generic issue evaluations. The review indicated that management attention is needed to improve uniformity and consistency in the licensing review and follow-up verification processes. Communication of expectations regarding documentation of informal communications with licensees is also needed.

1. INTRODUCTION

On December 4, 1995, the NRC received anonymous allegations regarding the Maine Yankee Atomic Power Station (Maine Yankee). It was alleged that Yankee Atomic Electric Company (YAEC), acting as agent for Maine Yankee Atomic Power Company (MYAPCo), the licensee for Maine Yankee, knowingly performed inadequate analyses of the emergency core cooling system (ECCS) and the containment to support applications for increases in the licensed maximum reactor power level.

When the NRC received the allegations, Maine Yankee had been shut down for refueling and steam generator repairs since February 6, 1995. In response to the technical issues raised by the allegations, the NRC sent an assessment team to YAEC headquarters in December 1995 to review and assess YAEC's small-break loss-of-coolant accident (SBLOCA) and containment peak-pressure analyses related to the power uprate amendments for Maine Yankee. On the basis of this assessment and information obtained during a subsequent meeting on December 18, 1995, the NRC issued on January 3, 1996, an order specifying requirements for reactor startup and limiting power to 2440 MWt. The order also specified requirements to be satisfied prior to an increase in power to the previously approved limit of 2700 MWt.

1.1 Historical Background

MYAPCo was originally granted a license to operate Maine Yankee at a power of 2440 MWt, based in part on a Combustion Engineering (CE) analysis of the ECCS. By application dated August 1, 1977, the licensee requested a single step increase in the maximum thermal power rating to 2630 MWt, based on the same CE ECCS model. On May 10, 1978, the NRC issued license amendment No. 38, which increased the licensed power level to 2630 MWt, but restricted operation to 2560 MWt until the Advisory Committee on Reactor Safeguards (ACRS) reviewed the power increase from 2560 to 2630 MWt. The power level of 2560 MWt is the level assumed in the Final Safety Analysis Report (FSAR) that supported the application for the operating license for Maine Yankee. On June 20, 1978, the NRC issued license amendment No. 39, which authorized the licensee to operate the plant at 2630 MWt.

On December 28, 1988, the licensee submitted a request to amend the license to increase the plant's maximum thermal power rating to 2700 MWt, again based on the same CE ECCS model. The NRC granted this amendment request on July 10, 1989, with license amendment No. 113.

Licensees are required by 10 CFR 50.46 to demonstrate that the ECCS would protect the core during a loss-of-coolant accident (LOCA). The ECCS cooling performance must be calculated for several postulated LOCAs of different sizes and other properties to provide assurance that the most severe postulated LOCAs are calculated. The regulation requires that the demonstration be performed using an acceptable evaluation model and allows two options: a conservative model meeting the criteria in Appendix K to Part 50, or a realistic model with an estimate of the uncertainty in the calculation. The second option was added in the 1988 amendment to the regulation because research has shown that Appendix K models are highly conservative.

Item II.K.3.30 of NUREG-0737, "Clarification of TMI Action Plan Requirements," which was issued in 1980, reflected a staff concern, in the wake of the TMI-2 accident, that a SBLOCA might be the most severe LOCA. This item called for licensees to either show that their current analyses were acceptable in light of the new concerns or submit for NRC approval revised SBLOCA models consistent with the requirements of Appendix K. The revised models were to account for comparisons with experimental data, including data from the LOFT and Semiscale facilities. Item II.K.3.31 stated that plant-specific small-break LOCA calculations, using models approved by the NRC under item II.K.3.30, should be submitted for NRC approval by all licensees to show compliance with 10 CFR 50.46, and that the calculations should be submitted within one year after staff approval of the LOCA analysis models.

In response to item II.K.3.30, YAEC submitted topical report YAEC-1300P, "RELAP5YA: A Computer Program for Light Water Reactor System Thermal-Hydraulic Analysis" dated October 1982. By letter dated October 14, 1988, the NRC forwarded its safety evaluation report (SER) on RELAP5YA to YAEC and found the topical report to be acceptable for referencing in licensing applications, under the conditions specified in the SER. By letter dated January 17, 1989, to MYAPCo, the NRC approved the application of RELAP5YA to Maine Yankee under the conditions specified in the staff's SER.

Item II.K.3.5 requested PWR licensees to study the need for automatic trip of reactor coolant pumps during LOCAs. Generic Letter 83-10, "Resolution of TMI Action Item II.K.3.5, Automatic Trip of Reactor Coolant Pumps," requested licensees to justify use of manual action to trip the reactor coolant pumps for SBLOCAs. In its reply dated June 28, 1985, the Maine Yankee licensee concluded that manually tripping the reactor coolant pumps was sufficient to satisfy the generic letter and 10 CFR 50.46. By letter dated April 15, 1986, the NRC accepted the licensee's position, which was based on analyses performed with RELAP5YA.

Item II.K.3.31 was closed for Maine Yankee by a letter to MYAPCo from the NRC project manager for Maine Yankee, dated May 8, 1989. The full text of the letter follows.

"By telecon May 5, 1989, Mr. S. Nichols of Maine Yankee informed me that your computer model and program you developed concerning Small Break LOCA analysis per NUREG-0737 II.K.3.30 and II.K.3.31 is operational and will be used to develop your next fuel reload (Cycle 12).

Since this constitutes implementation, this closes item II.K.3.31. Item II.K.3.30 was closed by letter dated January 30, 1989. Thus, this completes NRC review of your Small Break LOCA analysis. Your computer program, its verification and implementation may, in the future, be the subject of an inspection by NRC."

However, MYAPCo did not use RELAP5YA to develop Maine Yankee's Cycle 12 fuel reload. The application of RELAP5YA to Maine Yankee was not accomplished until 1993, well into Cycle 13, and was not submitted to the NRC for review. Furthermore, the approach that was used in the analysis was not consistent with the conditions stated in the staff's approval for RELAP5YA. Thus the licensee's analysis did not conform to the guidance in Item II.K.3.31.

1.2 Maine Yankee Lessons Learned Study and other Maine-Yankee-Related Activities

As a result of the allegations and the subsequent assessment by the NRC staff, the Office of Nuclear Reactor Regulation's (NRR) Associate Director for Technical Review (ADT), in a memorandum dated January 5, 1996, requested that the Director of NRR's Division of Systems Safety and Analysis (DSSA) address issues arising from the Maine Yankee experience. In response to ADT's request, the Maine Yankee Lessons Learned Task Group was formed, and in a memorandum dated April 25, 1996, the Director of DSSA provided a plan for addressing the issues. (The plan is included in this report as an appendix.) This report presents the results of that effort.

Other activities related to the Maine Yankee allegations were initiated in parallel with the Maine Yankee lessons-learned activity.

The Office of the Inspector General (OIG) has conducted an event inquiry into the NRC staff's actions related to regulation at Maine Yankee. On May 8, 1996, OIG provided its report on the event inquiry to the Commission. In the report, OIG arrived at the following nine conclusions:

1. That although the NRR staff reviewed the RELAP5YA computer code method, there was weakness in the review process. "OIG found that NRR reviewed the RELAP5YA code for Maine Yankee independent of the code plant specific application. Normally the codes and plant specific applications were reviewed together, but not so with the RELAP5YA code for Maine Yankee."

2. That the NRR staff was unaware of MYAPCo's non-compliance with five of the SER conditions until December 1995. "OIG found that the NRR staff did not follow-up on the SER with MYAPCo after the SER was issued in January 1989."

3. That the close-out of TMI Item II.K.3.31 by the NRC staff was inappropriate. "With respect to NRC's May 8, 1989, letter that closed out TMI Item II.K.3.31, OIG determined the following: the former Project Manager, the author of letter, did not follow the established practice of procedure of obtaining NRR technical staff review; the former Project Manager did not follow the established concurrence format of the PDI-3; the former Project Manager apparently made a unilateral decision regarding the closure of TMI Item II.K.3.31; and the former Project Manager was vague and made the close out letter subject to misinterpretation."

4. That there was a lack of NRR management oversight regarding the May 1989 letter. "OIG found that the NRR staff listed for distribution of the May letter had the opportunity to question the letter. OIG disclosed that none of the NRR staff listed for distribution questioned the letter or the close out of TMI Item II.K.3.31."

5. That the closeout of TMI Item II.K.3.31 was overlooked by the staff. "OIG learned that a review of the TMI Items status was conducted by NRR during 1989. The inquiry found that the NRR Project Managers, technical staff managers and senior officials had several opportunities to identify and resolve the erroneous closure of TMI Item II.K.3.31."

6. That the NRR Project Managers had several conversations with the licensee in which the licensee made commitments with respect to the SBLOCA analysis. "OIG found that the NRR Project Managers did not follow-up on the licensee's commitments."

7. That NRR staff did not have and presently does not have a formal licensing commitment tracking system. "OIG determined that the lack of a formal licensing commitment tracking system was a contributing factor to the NRC being unaware of MYAPCo's non-compliance with the SER and TMI Item II.K.3.31."

8. That NRR did not have a policy on Project Managers documenting and retaining their documentation of conversations with licensees. "One former Project Manager stated to OIG that he had a policy of not taking notes and if he did take notes, he would later destroy them. Another former Project Manager stated to OIG that any documentation of past conversations with the licensee would not have been retained by him. OIG found that the Project Managers' lack of documentation has placed the agency in a position of having to rely on the licensee's documentation when inquiring into past events."

9. That "the theme of NRR's reliance on the licensee resounded throughout this investigation. OIG revealed several examples where the NRR staff relied on MYAPCo to follow the NRC requirements and regulations, however, MYAPCo did not adhere to NRC requirements and regulations."

In a memorandum dated May 24, 1996, to the Executive Director for Operations (EDO), the Chairman requested the staff's plans to address the findings of the report. In a Commission paper dated June 21, 1996 (SECY-96-135), the EDO provided the staff's plans and stated that the staff would report the results and status of corrective actions taken and planned to the Commission by November 1996. Some of the issues raised by the OIG report were addressed in the Maine Yankee Lessons Learned Task Group plan.

The OIG report also found that MYAPCo did not report modifications to RELAP5YA and problems experienced with the code as required under 10 CFR

50.46, and found that the licensee did not use RELAP5YA in accordance with its SER and TMI Action Item II.K.3.31. In response to the OIG findings, and to respond to concerns by the Governor of Maine about the safety and the effectiveness of regulatory oversight of Maine Yankee, the Chairman initiated an independent safety assessment of Maine Yankee. The purpose of the independent safety assessment at Maine Yankee was to evaluate (1) the conformance of the plant to its design and licensing bases, (2) licensee self-assessments, corrective actions and improvement plans, (3) the root cause(s) of significant deficiencies, and (4) operational and overall plant safety performance. The report of the independent safety assessment was issued in October 1996.

2. CODE REVIEW PROCESS

It was determined that RELAP5YA was not applied for Maine Yankee in a manner conforming to the requirements of 10 CFR 50.46 and 10 CFR 50 Appendix K, nor had it been applied in a manner conforming to the conditions specified in the staff's SER, as necessary for NRC acceptance of the use of RELAP5YA for SBLOCA analyses for Maine Yankee. As a result, the staff issued an Order and Demand for Information restricting power to 2440 megawatts-thermal (MWt). This issue led to the formation of the Maine Yankee Lessons Learned Task Group to perform a self assessment of the code review process, the code modification process, staff followup of vendor and licensee code implementation, and to evaluate options for a catalogue of codes. The team has completed its assessment in these areas and the results and recommendations from this assessment are provided in the following discussion.

2.1 NRC Review and Approval of ECCS Methodologies

2.1.1 BACKGROUND

The review and approval process for an ECCS evaluation methodology is a complex technical effort, which can take a very long time to complete. The review process developed following lengthy hearings conducted in the early 1970's, and has evolved over the years. In order to put the information presented in later sections in context, a discussion of ECCS analysis methodologies, regulatory requirements, and the regulatory review process as it applies to ECCS evaluation models is provided below.

2.1.1.1 Staff-Approved ECCS Methodologies

While many licensees have developed their own in-house transient analysis capability, ECCS evaluation models have been almost exclusively developed by nuclear steam supply system (NSSS) and nuclear fuel vendors. The first evaluation models subject to extensive regulatory review were those developed under the 1971 interim acceptance criteria for ECCS evaluation models. These were large-break loss-of-coolant-accident (LBLOCA) methods developed by the four domestic NSSS vendors: Westinghouse (W), General Electric (GE), Combustion Engineering (CE), and Babcock and Wilcox (B&W). The LBLOCA was designated as the design basis accident, which determined the performance requirements of the ECCS. Under the design basis accident concept, the LBLOCA was the most limiting credible event. A system designed to mitigate this event was believed to be able to handle all other, less limiting events. Hence, it was thought that only this event needed to be considered. In conjunction with the design basis accident (DBA) concept, 10 CFR 50 Appendix K was developed to specify attributes of acceptable LBLOCA evaluation models (EMs), and 10 CFR 50.46 was incorporated to specify the acceptance criteria for ECCS based on analyses conducted using Appendix K codes. These criteria, which were promulgated in January of 1974, are applicable to all light water reactors which use zirconium fuel cladding. Each of the NSSS vendors submitted LBLOCA models for regulatory approval to comply with the final acceptance criteria.

For pressurized water reactors (PWRs), it was realized even before the final acceptance criteria were issued, that some of the required and acceptable features of ECCS evaluation models specified in Appendix K (notably those related to ECCS bypass and reflood) were not applicable for small breaks. Therefore, the PWR NSSS vendors developed and submitted separate models for analysis of SBLOCAs. For boiling water reactors (BWRs), there is significantly less change in phenomenology as break size decreases, so separate models have not been developed. Another reason is that BWRs have an automatic depressurization system (ADS), which opens valves to depressurize the reactor coolant system under accident conditions, allowing full ECCS flow to mitigate the accident consequences.

Following issuance of the final acceptance criteria, Exxon Nuclear Company, a supplier of both BWR and PWR fuel, developed ECCS methodologies for both types of plants. Exxon Nuclear later changed its name to Advanced Nuclear Fuels Corporation, and subsequently was acquired by and merged into Siemens Power Corporation. Within the last year, ABB-CE has obtained regulatory approval for a BWR ECCS methodology. This ECCS methodology development was initiated by Westinghouse and Asea Brown Boveri (ABB) in an attempt to enter the domestic BWR fuel market, and was completed by ABB-CE following ABB's acquisition of Combustion Engineering.

The Babcock and Wilcox Company has recently been acquired by Framatome and has changed its name to Framatome Technologies, Inc. (FTI). The FTI NSSS has a once through steam generator (OTSG) design, which differs from the U-tube steam generators used in Westinghouse and ABB-CE plants. The OTSG plants also have vent valves between the outlet plenum and the downcomer, which significantly changes many aspects of LOCA phenomenology. Therefore, a separate methodology is necessary to handle the OTSG plants. The original ECCS method developed by B&W applied only for OTSG plants. When B&W began refueling U-tube plants, it was necessary to develop a separate methodology. Therefore, FTI now has separate methods for OTSG and U-tube steam generator plants.

Table 1 summarizes approved ECCS models developed by NSSS and fuel vendors.

Table 1. Summary of NSSS and Fuel Vendor ECCS Methodologies

VENDOR	METHODOLOGY (CODES)
Westinghouse Electric Corp.	PWR LBLOCA (BART, BASH, SATAN, etc.)

Westinghouse Electric Corp.	PWR SBLOCA (NOTRUMP)
General Electric Co.	BWR LOCA (SAFE/FLOOD)
General Electric Co.	BWR LOCA (SAFER/GESTR)
ABB-Combustion Engineering	PWR LBLOCA (CEFLASH-4A, etc.)
ABB-Combustion Engineering	PWR SBLOCA (FLASH-4AS, etc.)
ABB-Combustion Engineering	BWR LOCA (GOBLIN, etc.)
Framatome Technologies, Inc.	OTSG PWR LBLOCA (CRAFT2, etc.)
Framatome Technologies, Inc.	OTSG PWR SBLOCA (CRAFT2, etc.)
Framatome Technologies, Inc.	RSG PWR LBLOCA (RELAP5/MOD2, etc.)
Framatome Technologies, Inc.	RSG PWR SBLOCA (RELAP5/MOD2, etc.)
Siemens Power Corp.	PWR LBLOCA (RELAP4-EM, TOODEE2, etc.)
Siemens Power Corp.	PWR SBLOCA (ANF-RELAP, TOODEE2, etc.)
Siemens Power Corp.	BWR LOCA (RELAP4/ENC-28B, etc.)

Some utilities have developed or attempted to develop their own in-house LOCA methods. In addition to YAEC, which has approved LOCA methodologies for PWRs and BWRs, the only other utility which has an approved LOCA methodology is Northeast Utilities, which has an approved PWR SBLOCA method for the Haddam Neck Plant. Other utilities have taken the approach of performing their own LOCA analysis by using methodologies already developed by a vendor. Utilities performing their own ECCS analyses using vendor methods include Virginia Power, which uses Westinghouse methods, and Texas Utilities, which uses Siemens methods. Texas Utilities submitted Topical Reports which document the application of Siemens methods to Comanche Peak for SBLOCA and LBLOCA. The Northeast Utilities, YAEC, FTI, and Siemens methodologies are all based in part on versions of the RELAP code, which is an NRC-developed code.

2.1.1.2 ECCS Methodologies

ECCS methodologies are based on one or more computer codes, which are used in parallel or in sequence to simulate the complete LOCA event. The number of codes used varies among the vendors and also varies for the type of event. Early PWR methods tended to use separate codes for the blowdown portion and the reflood portion of the event, and also to use a separate code for the fuel heatup or hot channel analysis. Separate codes are generally also used to determine the fuel temperature initial conditions and the containment response. These codes were not examined in detail for this report. Vendors may include a description of the fuel performance and containment codes along with the base methodology. If such descriptions are not included, then separate Topical Reports are submitted to describe these codes.

The process of obtaining regulatory approval for an ECCS evaluation generally requires three steps. First, topical reports describing the computer codes to be used are submitted for approval. These reports may include benchmarking of the codes against relevant experimental data. Then a second topical report is submitted describing the ECCS analysis methodology. This so-called "Applications Report" describes how the computer codes will be used together to perform the complete ECCS evaluation. Sample calculations are usually included to illustrate the application. Finally, plant specific analyses, or analyses specific to groups of plants with similar designs (plant class specific), are performed and documented in a third report, the licensing analysis. This report actually determines the limiting conditions for operation (LCOs) that result from the LOCA analysis.

Utilities performing their own analyses using an approved vendor method have submitted only the licensing analysis report, which may contain some aspects of the applications report, e.g., a noding study for the plant specific application.

2.1.1.3 ECCS Model Revisions and Replacements

For various reasons, NSSS and fuel vendors have occasionally updated their ECCS models, either making revisions to the codes and/or methodology to improve model accuracy, or developing a completely new methodology based on different computer codes. 10 CFR 50.46 allows changes to or corrections in existing models without prior NRC approval. Such changes must be reported to the staff at least annually. External factors have sometimes influenced the updating and development of revised ECCS models. Three factors which influenced revisions or new ECCS models are:

1. NUREG-0737, Post TMI Action Items, in particular item II.K.3.30, justification of SBLOCA methods.
2. SECY-83-472, which provided an alternative ECCS analysis approach for performing LBLOCA analyses in conformance with 10 CFR 50 Appendix K. The paper also included a discussion of the basis for the revision to 10 CFR 50.46 which allowed realistic analyses.
3. Revision of 10 CFR 50.46 to allow realistic LOCA analysis.

Following the TMI accident, the NRC requested each licensee to justify the SBLOCA methods applied to its plant. NUREG-0737, in particular item II.K.3.30, resulted in some of the SBLOCA methodologies being revised, and/or additional justification being submitted to demonstrate conservatism.

Extensive confirmatory research conducted by the NRC in conjunction with industry on ECCS performance (e.g., LOFT, Semiscale, FLECHT, and MIST) and associated NRC code development (RELAP, TRAC, etc.) demonstrated that the evaluation models for LBLOCA were conservative and embodied considerable margin in terms of the predicted peak clad temperature (PCT). Some relief, within the then-current Appendix K, was granted by SECY-83-472. This prompted revisions of LOCA models. Some of these revisions demonstrated additional margin in the ECCS and were used to justify power uprate requests.

Regulations were promulgated in 1988 in revisions to 10 CFR 50.46, allowing for realistic LOCA analysis. Statistical uncertainty evaluation was required to quantify margins. This relief was optional, in the sense that continued use of the old Appendix K models was permitted, subject to some restrictions on specific models (e.g., Dougall-Rohsenow post-CHF heat transfer). Regulatory Guide 1.157, dated May 1989, was issued to provide guidance for realistic calculations of ECCS performance. Only one methodology has been approved under these provisions, Westinghouse realistic LBLOCA using WCOBRA-TRAC. Combustion Engineering, Siemens, and General Electric have developed realistic methodologies which have not yet been reviewed or approved by the NRC. The WCOBRA-TRAC methodology was the subject of extensive regulatory review, including lengthy review by the Advisory Committee on Reactor Safeguards (ACRS).

2.1.2 EXAMINATION OF NRC REVIEW AND APPROVAL OF ECCS METHODOLOGIES

The Maine Yankee lessons learned task group, with the assistance of its contractor, Scientech, Inc., has examined previous NRC reviews leading to the acceptance of several ECCS evaluation models and model revisions to assess the review process. These include the 14 NSSS and fuel vendor methodologies listed in Table 1. Also examined were YAEC's LOCA methodologies, the SBLOCA methodology developed by Northeast Utilities for the Haddam Neck Plant, and the application of the Siemens small break and LBLOCA methodologies by Texas Utilities Electric for the Comanche Peak plants.

The examination covered all available documentation of the reviews, including submittals, staff requests for additional information (RAIs), responses to RAIs, meeting summaries, and staff SERs, which were gathered from both publicly available and internal NRC document archives. The information extracted from the documents includes: computer codes which are used for the analyses, number and description of sensitivity cases, number and description of benchmark cases, sample applications, number and level of detail of the RAIs, meetings held including presentation to the ACRS, number and content of conditions imposed in the staff SERs, and availability of verification of the implementation of the SER conditions.

2.1.2.1 Results of Examination

Difficulties were encountered in retrieving some of the documentation of staff reviews of ECCS methodologies. The amount of information that could be retrieved was strongly dependent on the time frame in which the review was performed. Many of the reviews were initiated and/or completed before the NRC's document retrieval system, NUDOCS, was developed. For more recent reviews, documentation was readily available.

A great diversity in the content and organization of the ECCS methodology documentation was evident in this examination. Unlike the FSAR, there is no format and content guide for code and methodology topical reports. Each vendor or licensee exhibits individual style, organization, and content. In general, there are three types of topical reports associated with ECCS methodologies. The first type describes one or more of the computer codes used to perform the analyses. The second type describes the methodology and is called the applications report. The third type documents the plant or plant class specific analysis. However, the documentation examined shows that there are many exceptions to this format and a wide variation in terms of the breakdown of information between the various types of topical reports. One of the vendors has made a practice of including the SERs for previous revisions of their topical reports in their submittals. This was found to be helpful in reconstructing the review process for their methodologies.

NUREG-0390, "Topical Report Review Status," which is published semiannually, provides industry with procedures for submitting topical reports; guidance on how the NRC processes and responds to topical report submittals; and an accounting, with review schedules, of all topical reports currently accepted for review by the NRC. Beginning in 1984, the NUREG guidance includes procedures for publication of approved versions of reports. The current guidance states that, after the NRC accepts a licensing topical report for referencing, the sponsoring organization should publish an approved version and that a copy of NRC's transmittal letter and its evaluation report should be inserted immediately after the title page of the approved version. In early documentation (prior to the mid 1980's), considerable difficulty was encountered in locating approved versions of the topical reports, i.e. with the SER included.

For most of the methodologies examined, the team was able to retrieve the RAIs and responses to RAIs. The RAIs were numerous and detailed, indicating an adequate scope and depth of review. Among the documentation examined, there was wide variation in whether and how RAI responses were included in the approved versions of the topical reports. Documentation of the RAI responses was included as appendices, separately bound supplements, in a separate licensing documents volume, or not included at all. Early period documentation often does not include the RAI responses. NUREG-0390 does not provide guidance with regard to including RAIs and responses to RAIs in the approved versions of topical reports.

In addition to the recent problems with the RELAP5YA code, inadequacies have been discovered in the Siemens LBLOCA model. Sample applications were not provided at the time of review of either of these methodologies, which may have contributed to the problems that have been encountered with these methodologies. Sample applications were provided for all other ECCS methodologies examined. Benchmark cases were provided for all of the methodologies examined, and with the exception of one methodology for which documentation was limited, sensitivity cases were provided for all of the methodologies examined.

There was wide variation in the number and content of conditions or restrictions imposed in the staff's SERs. Most of the SERs contained some conditions or restrictions, and a few of the SERs contained 9 or more. It was found for some of the topical reports that the conditions imposed in the staff's SER were not consistent with the conditions in the attached technical evaluation report (TER) prepared for the staff by a contractor. This inconsistency could cause confusion and potentially result in the improper implementation of the conditions. There was no information available in the documentation examined to indicate that the staff had verified vendor/licensee conformance to the restrictions and/or conditions in the staff's SERs. Verification of conformance to SER conditions is discussed further in sections 2.2.2 and 2.2.3.

As a result of the Maine Yankee issue, the regulatory status, i.e. enforceability, of conditions imposed in staff SERs has been called into question. Therefore, the technical staff has been instructed that conditions should no longer be imposed in staff SERs approving topical reports or technical specification (TS) amendments. Alternatively, licensees and vendors must commit to any restrictions or conditions which must be satisfied for staff approval of topical reports or TS amendments. Any licensee or vendor commitments which are relied upon for staff conclusions will be specified in the letter granting approval of topical reports or TS amendments and in the staff SER supporting approval. It should be noted that recent guidance from the Office of General Counsel (OGC) emphasizes the lack of legal force of vendor and licensee commitments without imposition by order or incorporation into a license.

ACRS involvement was evident in the reviews of about half of the ECCS methodologies examined.

2.1.3 RECOMMENDATIONS FOR NRC CODE REVIEW PROCESS

Based on the team's review, the following recommendations have been developed:

(1) **A standard format and content guide for topical reports documenting ECCS methodologies should be developed.** The guide would provide assurance of uniformity and consistency in the level of documentation and validation that is provided.

(2) **The guidance of NUREG-0390 should be modified to specify that RAIs and responses to RAIs be included in the approved versions of topical reports to provide complete documentation of the staff's review.**

(3) **Sample applications of codes and methodologies should be required to be submitted for approval of codes and methodologies.**

(4) **Conditions or restrictions should not be imposed in topical report safety evaluations without prior commitment by the vendor or licensee. (Even with prior vendor or licensee commitment, conditions and restrictions on topical report approvals should be minimized.**)

(5) **Clear guidance should be provided to the staff regarding the regulatory status and enforceability of licensee and vendor commitments.**

2.2 Followup on Licensee/Vendor Implementation

The Maine Yankee Lessons Learned Task Group was directed to review the results of completed audits and inspections in this area, to determine if the Core Performance Action Plan should be modified in light of its findings to include more emphasis in the area of licensee/vendor code implementation with an emphasis on how SER conditions are satisfied, and to determine if the guidance in Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits from Technical Specification," needs to be modified to be more specific with regard to SER conditions.

2.2.1 CORE PERFORMANCE ACTION PLAN

The Core Performance Action Plan was developed in 1994 to assess the impact of reload core design activities on plant safety. The action plan includes several major task areas including the following, which are relevant to followup of code implementation: (1) inspection of nuclear fuel vendors; (2) inspections of licensees' reload analyses; and (3) evaluation of inspection guidance.

Since 1994, the staff has completed five performance-based vendor inspections and four licensee reload analyses inspections. Through these inspection efforts, a total of 26 plant specific cycle reload analyses have been evaluated. The scope of these inspections has included the reload analysis methods and computer code models. The bases for inspection of code implementation include 10 CFR 50 Appendix B and the appropriate American Nuclear Society (ANS) and Institute of Electrical and Electronics Engineers (IEEE) code documentation, validation, and verification standards. Compliance with applicable SER restrictions is also reviewed.

To more effectively evaluate licensee reload analysis capability, the scope of inspection activity was expanded for fiscal year 1997. The planned inspection activities will emphasize vendor and licensee interaction and oversight and will include: six licensee reload inspections (focus on in-house analyses), four licensee/vendor interface specific inspections, four reactive or issue followup inspections, one direct vendor inspection, and four vendor lead test assembly program inspections.

The team has reviewed the results of completed inspections to determine the extent to which inspections have emphasized code implementation and conformance with the conditions of staff SERs. Areas of concern that have been identified by the Core Performance Action Plan include organizational interfaces, data transfer control quality assurance (QA), and potential weaknesses of oversight audit functions conducted by licensee technical staff. While there has been significant emphasis in the completed inspections on reviewing the implementation of physics codes, core thermal-hydraulic codes, fuel performance codes, and non-LOCA transient and accident codes to perform reload core design and safety analyses, there has not been an emphasis in the area of implementation of ECCS evaluation methodologies, nor has it been documented in inspection reports that conformance with SER conditions has been verified for the codes that were reviewed.

While there is a clear regulatory requirement that ECCS analyses be performed using an NRC approved methodology (i.e., 10 CFR 50.46), the team noted that the regulatory basis for staff review and approval of other codes (e.g., physics codes, core thermal-hydraulic codes, fuel performance codes, and non-LOCA transient and accident codes) used to perform reload core design and safety analyses is not clear. Historically, vendors and licensees have submitted such codes for NRC review and approval.

An inspection is planned for early 1997 at Siemens Power Corporation (SPC), focusing in part on SPC's various analytical methodologies. This aspect of the inspection was motivated by problems identified in a reflood heat transfer correlation in SPC's LBLOCA evaluation model. The inspection will also emphasize other areas, which are typically covered by the Core Performance Action Plan inspections.

2.2.2 CORE OPERATING LIMITS REPORTS

Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits from Technical Specifications," was issued to describe and provide guidance for the implementation of an alternative to specifying the values of cycle-specific parameter limits in the TSs. This alternative approach, which has been adopted by most licensees, eliminates the majority of TS amendments associated with changes in the values of cycle-specific parameters in TSs. Under this approach, the TSs specify the methodologies that the licensee will use to calculate cycle-specific parameters. The actual values of the parameters are maintained in a core operating limits report (COLR).

While the guidance in Generic Letter 88-16 specifies that cycle-specific parameter limits should be calculated using NRC approved methodologies, it is silent with respect to assurance that SER conditions are satisfied for the methodologies that are used to calculate the cycle-specific parameter limits. A sample of staff SERs for TS amendments for individual licensee conversions to the GL 88-16 approach were also examined and were silent with regard to conditions in the staff's SERs approving methodologies, but did note that all of the referenced methodologies had been previously reviewed and approved by the NRC.

2.2.3 RECOMMENDATIONS FOR FOLLOWUP ON LICENSEE/VENDOR IMPLEMENTATION

Based on the team's review, the following recommendations have been developed:

(1) **Additional emphasis should be placed on audits and inspections of implementation of ECCS methodologies.**

(2) **Additional emphasis should be placed on verification of conformance to SER conditions in the area of inspections and audits.**

(3) **The Core Performance Action Plan should be modified to incorporate recommendations 1 and 2 above.**

(4) **The regulatory basis of staff review and approval of codes other than ECCS methodologies should be examined, and consideration should be given to revising the regulations to address this issue.**

(5) **Guidance should be provided to the staff to ensure that conformance to topical report SER conditions is verified whenever licensee or vendor application of codes or methodologies is under staff review, (e.g., plant specific applications of approved vendor methodologies, power uprate applications, TS amendments, and COLR conversions.)**

2.3 Catalogue of Codes

2.3.1 DISCUSSION

During its review of recent events involving the Maine Yankee SBLOCA analysis and other code related issues, the staff has found it necessary to identify and understand the review and approval process for a particular code or analysis, as well as the limits of its applicability. In addition, the staff needs to know the status of codes when it is reviewing license amendment submittals that employ analytical codes. However, the staff does not currently maintain any coordinated list of the review and approval of analytical methods; therefore, the memory of staff members and the ability to retrieve documentation from the docket file or NUDOCS must be relied upon to answer subsequent questions about a particular model or method. Significant difficulties were encountered while obtaining documentation to support the examination of the LOCA code review process discussed in Section 2.1.2.1. The staff response to code related events could be enhanced, and the potential for code errors and misapplications would be reduced, if the staff maintained a catalogue of analytical code reviews and approvals.

2.3.2 RECOMMENDATION

A catalogue of codes should be developed and be updated continuously. Its scope should at least include thermal-hydraulic codes, nuclear physics codes, nuclear fuels codes, fuel/mechanical codes, and post-LOCA containment performance codes.

A statement of work has been drafted which describes the actions that need to be performed by a contractor to develop, install, and load the initial set of data into an NRR code catalogue. Initially, the planned scope of the catalogue will include thermal-hydraulic codes, nuclear physics codes, nuclear fuels codes, and fuel/mechanical codes, which are used to understand the performance of the reactor core and reactor coolant system during transients and accidents, and also post-LOCA containment performance codes. Optionally, the scope may be expanded to cover other codes such as structural analysis codes, radiological protection and shielding codes, mechanical engineering codes, and electrical system codes. The catalog would be updated by the NRC staff.

3. POWER UPRATE REVIEW PROCESS

The allegations on Maine Yankee included the assertion that inadequate analyses were performed by YAEC to support two power uprates (license amendments to increase the plant's licensed maximum reactor power level) for Maine Yankee. This assertion implies that the staff's reviews of the Maine Yankee power uprate applications were inadequate. To understand the adequacy of the staff's power uprate review process, the Maine Yankee Lessons Learned Task Group, with the assistance of its contractor, Scientech, Inc., examined past power uprate reviews and assessed them against current procedures and practice.

On the basis of an inquiry to operating reactor project managers and a search of NUDOCS, thirty-one completed power uprates were identified. Some of these uprates had not required a new safety review by the staff, because the FSARs supporting the applications for operating licenses for the plants had been written assuming the power levels to which the plants were later uprated, and the staff safety evaluations of these applications had been performed assuming these power levels. For each of these early plants, the Commission had responded to the application for an operating license by granting a provisional operating license, which limited the thermal power output of the reactor until its performance could be further evaluated. Later, the Commission granted a full-term operating license at the requested power level. Of the thirty-one completed power uprates identified, nine are in this category. In each of these cases, the staff reviewers verified that all the analyses in the FSAR were performed at the power level requested in the uprate application, re-reviewed any re-analyses performed by the uprate applicant, and confirmed that the plant was now fully analyzed at the proposed power level and core loading.

The remaining twenty-two power uprate applications, which are listed in Table 2, required supplementary reviews by the staff. These uprate reviews were assessed by the team.

Table 2. Power Uprates That Required Supplementary Reviews

PLANT	DATE
Callaway	03/30/88
Calvert Cliffs 1	09/02/77
Calvert Cliffs 2	10/19/77
D.C. Cook 2	01/14/83
Fermi 2	09/09/92
Fort Calhoun	08/15/80
Hatch	08/31/95
Limerick 1	01/24/96
Limerick 2	02/16/95
Maine Yankee	05/10/78
Maine Yankee	07/10/89
Millstone 2	05/12/79
Nine Mile Point 2	04/28/95
North Anna	08/25/86
Peach Bottom 2/3	10/18/94
St. Lucie 1	11/23/81
St. Lucie 2	03/01/85
Surry	08/03/95
Susquehanna	04/11/94
Vogtle	03/22/93
Wolf Creek	11/10/93
WNP-2	04/25/95

In each case, the team reviewed the staff SER, the power uprate application, and the licensee responses to staff requests for additional information. Each power uprate review was assessed in light of the Maine Yankee allegations and the problems the staff found in its December 1995 assessment of YAEC's SBLOCA and containment peak-pressure analyses for Maine Yankee(1). The team looked for the following attributes in the staff's review: (1) Did the reviewer determine whether the analytical codes used to support the uprate application were the same as those which were approved and whether the codes were approved for use for the purpose to which they were being applied? (2) If the licensee was applying the code to the subject plant for the first time, did the reviewer confirm that the code was used in accordance with conditions in the SER that approved the code? (3) Did the reviewer review all analyses beyond those that were approved? (4) Were technological developments and plant modifications that had occurred since approval of the analytical codes used by the applicant appropriately considered? (5) Did the reviewer establish the license conditions and technical specification changes needed for the license amendment? (6) Did the reviewer address what follow up is needed to verify that uprate changes are incorporated into the updated final safety analysis report (UFSAR)?

In addition, the team compared the scope of the uprate reviews to identify differences in the safety topics considered in the reviews.

3.1 GE BWR Power Uprate Program

The NRC does not have a standard procedure for reviewing power uprate applications, but the staff has taken a generic position on a GE program for uprating BWRs. In December 1990, GE Nuclear Energy submitted a licensing topical report entitled "Generic Guidelines for General Electric Boiling Water Reactor Power Uprate." The original proposal by GE did not limit the power increase that a licensee could request through the program. A June 1991 revision to the topical report contained proposed licensing criteria for review topics such as transient analyses, containment response, ECCS/LOCA response, and the set point methodologies for instrumentation and control; and GE agreed to limit the maximum power increase to be made available under the program to approximately 5 percent of rated thermal power. An increase of this magnitude or less is called a "stretch power" increase, because the nuclear steam supply system was designed for this higher power level, although the vendor-guaranteed or "nameplate" rating of the reactor was the power level considered in the FSAR. In September 1991, the staff sent to GE Nuclear Energy the staff's position paper regarding the proposed generic power uprate program, and in December 1991, the staff sent the position paper to the Commission. The position paper affirmed the staff's commitment to developing the program and included a conditional approval of GE's topical report. In July 1991, GE Nuclear Energy submitted a licensing topical report entitled "Generic Evaluations of General Electric Boiling Water Reactor Power Uprate," which provided generic bounding analyses and equipment evaluations, and in July 1992, the staff sent to GE Nuclear Energy the staff's SER for the topical report. The staff found that the topical report, when combined with sufficient additional plant-specific information, constitutes an adequate basis for the review of individual licensee power uprate requests. The staff position paper and the SER on GE's BWR power uprate generic analyses prescribe the scope, depth, methodology, criteria, and limitations of a BWR stretch power uprate review in the GE power uprate program.

The lead plant for the GE BWR generic power uprate approach was Fermi 2. The staff review satisfied some of the criteria identified earlier in this section. The staff position paper on the proposed generic power uprate program addressed selection and approval status of codes used to support the uprate application, and the position paper placed two conditions on the proposed use of two new codes in the generic power uprate program. SAFER/GESTR (for ECCS analysis) is approved; if it is used, a baseline run at the present power level must be included so that the true effect of the power uprate can be assessed. SHEX (for calculation of suppression pool response to LOCA) is not approved for generic use; if used, use must be justified. These codes were used for Fermi 2, and the SER showed evidence that the conditions in the position paper were satisfied. The review also included consideration of technological developments and plant modifications that had occurred. Design changes were reviewed, and the code used for SBLOCA analysis of the ECCS was in conformance with Item II.K.3.31 of the TMI Action Plan. The review also considered at least most changes needed in the plant's technical specifications, although some technical specification changes (e.g. changes to TS 3.8.4.3, which requires thermal overload protection for MOVs) were not discussed in the SER. The ACRS conducted a limited review of the power uprate for Fermi 2.

The team did, however, find room for improvement in the Fermi 2 review with respect to the identified criteria:

1. The power uprate SER did not confirm that use of new (i.e., not previously approved for use at that plant) codes was in accordance with conditions in the SERs that approved the codes.

2. With a few exceptions, it was not clear from the uprate application or the SER which analyses were done with codes that were previously used and approved for that plant and which were not.

3. The SER did not confirm that new codes used in the uprate analyses were identified in the proposed technical specifications.

4. The review did not consider changes needed to the plant's updated final safety analysis report (UFSAR). The licensee's application for power uprate was supported by much design analysis, which included the use of new analytical codes. The staff should ensure that this new information is incorporated into the UFSAR.

   Applications for construction permits and operating licenses are accompanied by safety analysis reports, so staff reviews of these applications need not be concerned with this problem. By referring to the FSAR for the plant, the license makes the FSAR part of the licensing basis. But a power uprate application is for a plant that already has a current plant safety analysis report, the UFSAR. The supporting material for the power uprate application is submitted as ad hoc reports and responses to staff requests for additional information. Some of the supporting analyses may have already been submitted, perhaps only for information, in such forms as a core operating limits report for the most recent core reload. If the license amendment does not specify the new analyses and codes, they do not become part of the licensing basis. Mentioning them in the SER is not enough.

These observations are not unique to Fermi 2. They are common to all completed uprate reviews. Furthermore, because of the standard nature of the reviews in the GE power uprate program, it is less likely that observations 1 and 2 are indicative of actual omissions in these reviews than in the other power uprate reviews.

The December 1990 GE topical report described the methods to be used in uprate analyses. The team found, however, that the documentation of the individual plant uprate reviews in the GE power uprate program did not always document confirmation that the methods described in the topical report were those used in the plant specific uprate.

The scope of the Fermi 2 power uprate review was quite extensive. The team found only two review areas included in some other uprate reviews that were not addressed in the Fermi 2 review (except of course for PWR-specific areas):

(1) human factors and (2) the impact of the uprate on General Design Criterion (GDC) 17, Electric Power Systems, considerations (10 CFR Part 50, Appendix A). Consideration of human factors was included in the Hatch and Nine Mile Point reviews. Consideration of GDC 17 was included in the submittals that were based on the Westinghouse methodology described in Westinghouse report WCAP-10263, "A Review Plan for Uprating the Licensed Power of a Pressurized Water Reactor Power Plant," January 1983. The Fermi 2 review, however, did include station blackout. The list of contributors to the Fermi 2 SER did not include reviewers in the Human Factors Assessment Branch or Electrical Engineering Branch.

Seven other uprates used the GE BWR generic power uprate approach: Hatch, Limerick 1, Limerick 2, Nine Mile Point 2, Peach Bottom, Susquehanna, and

WNP-2. With the following exceptions, all of these uprate reviews shared the strengths and weaknesses of the Fermi 2 review. The plant-specific evaluation of SHEX that was prescribed by the staff position paper on the GE BWR power uprate program apparently was not provided for Limerick 1 or Limerick 2. The Nine Mile Point 2 review did not include the standby liquid control system (SLC). The Hatch and Nine Mile Point reviews included human factors.

The SERs for the uprates that used the GE BWR generic power uprate approach listed the reviewers that contributed to the SERs. None of the SERs listed an electrical reviewer. For Fermi 2, Limerick 1, Limerick 2, Peach Bottom, Susquehanna, and WNP-2, a human factors reviewer was not listed. For Susquehanna, a radiological reviewer was not listed.

3.2 Other Completed Power Uprates

All of the other power uprates the team examined were approximately 5 percent or less, similar to those in the GE BWR power uprate program.

Four of the uprates used the Westinghouse methodology from WCAP-10263: Callaway, North Anna, Surry, and Vogtle. In addition, the scope of the Wolf Creek submittal seems to have been modeled after the Westinghouse methodology. In the Surry review, the codes used for ECCS, transient, and containment response analyses were identified, and their approval status was reviewed. For other codes, however, it was not clear from the uprate application or the SER which analyses were done with codes that were previously used and approved for that plant and which were not. In the Callaway, North Anna, Vogtle, and Wolf Creek reviews, this statement is also true of the codes used for ECCS, transient, and containment response analyses. In all cases, the power uprate SERs did not confirm that use of new codes was in accordance with conditions in the SERs that approved the codes. The reviews, in all cases, did not show evidence that they included consideration of changes, in knowledge and in the plant, that had occurred since the analytical codes used by the applicant were approved. The reviews also did not consider changes needed to the plant's updated final safety analysis report (UFSAR).

The scope of the Callaway and Surry reviews was the same as the scope of the Fermi 2 review (disregarding PWR- or BWR-specific review areas), except that the Callaway and Surry reviews included consideration of GDC 17 and did not include consideration of station blackout. The review areas not addressed in the scopes of the North Anna, Vogtle, and Wolf Creek reviews are indicated in Table 3, which lists review areas not addressed for all twenty-two uprate reviews.

Table 3. Review Areas Not Addressed in Uprate Reviews

(BWRs are compared to Fermi 2 + GDC 17 + human factors, and PWRs are compared to Surry + SBO + human factors.)

PLANT	NSSS	REVIEW AREAS NOT ADDRESSED
Callaway[2]	W	SBO human factors
Calvert Cliffs 1	CE	reactor vessel/internals stresses CRD mechanisms SG tube integrity RCPs pressurizer piping GDC 17 SBO EQ fire protection control room habitability LOCA/MSLB containment performance safety-related pumps NPSH post-LOCA combustible gas control service water CCW SFP cooling HVAC human factors
Calvert Cliffs 2	CE	(same as for Unit 1)
D.C. Cook 2	W	reactor vessel/internals stresses CRD mechanisms SG tube integrity RCPs pressurizer piping SBO EQ fire protection control room habitability

		safety-related pumps NPSH service water CCW SFP cooling HVAC radwaste circulating water system main steam human factors
Fermi 2[1]	GE	GDC 17 human factors
Fort Calhoun	CE	reactor vessel/internals stresses CRD mechanisms SG tube integrity RCPs pressurizer piping LOCA/MSLB containment performance safety-related pumps NPSH post-LOCA combustible gas control service water CCW SFP cooling HVAC EQ fire protection circulating water system main steam main turbine SBO human factors
Hatch[1]	GE	GDC 17
Limerick 1[1]	GE	GDC 17 human factors
Limerick 2[1]	GE	GDC 17 human factors
Maine Yankee 1978	CE	GDC 17 SBO safety-related pumps NPSH post-LOCA combustible gas control I&C setpoints human factors (Systems, structures, & components were not specified; merely general statement that staff evaluated effect of uprate on structures & systems)
Maine Yankee 1989	CE	reactor vessel/internals stresses CRD mechanisms SG tube integrity RCPs pressurizer piping GDC 17 SBO safety-related pumps NPSH post-LOCA combustible gas control service water system I&C setpoints SFP cooling HVAC fire protection circulating water system main steam main turbine control room habitability human factors
Millstone 2	CE	reactor vessel/internals stresses CRD mechanisms

		RCPs pressurizer GDC 17 SBO MSLB containment performance safety-related pumps NPSH post-LOCA combustible gas control service water system CCW SFP cooling HVAC fire protection circulating water system main steam main turbine human factors
Nine Mile Point 2[1]	GE	GDC 17 SLC
North Anna[2]	W	post-LOCA combustible gas control HVAC fire protection SBO human factors
Peach Bottom 2/3[1]	GE	GDC 17 human factors
St. Lucie 1	CE	RCS flow AFW RHR reactor vessel/internals stresses CRD mechanisms SG tube integrity RCPs pressurizer piping LOCA/MSLB containment performance safety-related pumps NPSH post-LOCA combustible gas control service water CCW SFP cooling HVAC EQ fire protection GDC 17 SBO I&C setpoints radwaste systems human factors
St. Lucie 2	CE	(same as for Unit 1)
Surry[2]	W	SBO human factors
Susquehanna[1]	GE	GDC 17 human factors
Vogtle[2]	W	fire protection SBO human factors
Wolf Creek[2]	W	safety-related pumps NPSH human factors
WNP-2[1]	W	GDC 17 human factors
Notes: [1]This uprate followed the GE BWR uprate approach. 2		

This uprate followed the Westinghouse uprate approach or scope.

The remaining uprate reviews were earlier uprate reviews and did not follow a generic uprate methodology established in a topical report. Compared to the reviews discussed above, they were quite limited in scope, as can be seen from Table 3.

Some of the variability between reviews in the areas covered may be a result of inconsistent application of the assumed reactor power by different reviewers during the original reactor licensing review. If an area was not addressed in the uprate SER or RAI, it was listed as not addressed in the review. Many reviewers, particularly in the DBA consequence reviews, assumed the stretch power level (105% of maximum power) while others used the guidance in Reg. Guide 1.49 (102% of maximum power). Therefore, in a particular area, the uprated power may have been determined to be within the original licensing basis. In these cases a reanalysis would not be necessary. However, if a technical area was not re-reviewed because the current licensing basis was still bounding, the power uprate SERs should have clearly documented it.

With one exception, the remaining uprate reviews did not appear to satisfy the criteria identified earlier in this section. The exception is that in the 1989 Maine Yankee review, the reviewer assessed analyses that used unapproved methods, although the reviewer did not mention that the SBLOCA analysis was not done using the recently approved SBLOCA code RELAP5YA.

A noteworthy aspect of the Millstone review was the cover letter for the power uprate amendment. This letter documented the staff's finding that certain licensee actions were needed and indicated that the licensee's staff had agreed to supply documentation of these actions on the schedule indicated in the letter. (The license amendment itself changes only the technical specifications and the paragraph in the license that refers to the technical specifications.) This is an example of staff reliance on informal licensee communication loosely tied to an "informal licensee commitment."

3.3 GE BWR Extended Power Uprate Program

In February 1995 GE Nuclear Energy submitted a licensing topical report entitled "Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate." The report contained proposed guidelines to be followed by BWR licensees in preparing requests to increase the thermal power up to 120 percent of the original licensed thermal power. In February 1996 the staff sent to GE Nuclear Energy the staff's position paper regarding the proposed extended power uprate program. The position paper presented the staff's position concerning (1) the expected content of individual licensee submittals, (2) the scope and depth of review topics, (3) approved methodologies to be used in evaluating selected review topics, (4) licensing approach and criteria for extended power uprate, (5) expected review schedules for individual licensee amendment requests, and (6) specific limitations to the use of the power uprate program.

In March 1996, GE Nuclear Energy submitted a licensing topical report entitled "Generic Evaluations of General Electric Boiling Water Reactor Extended Power Uprate," which provided generic bounding analyses and equipment evaluations. This topical report and the one issued in February 1995 both incorporated the lessons learned from the previous stretch power uprate program. In July 1996, Monticello, the lead plant for the BWR extended power uprate program, submitted its application for a power uprate of 6.3 percent. The staff is currently reviewing both the Monticello application and GE's March 1996 topical report.

Under the NRR Task Action Plan for the BWR Extended Power Uprate Program established in April 1995, the staff plans to develop a Standard Review Procedure for BWR uprates. This Standard Review Procedure will be based on (1) the staff position paper and the SER on GE's BWR extended power uprate generic analyses, and (2) the SER on the lead plant. The Standard Review Procedure will be applicable for both extended and stretch power uprates, with a few minor differences.

3.4 Conclusions

Examination of power uprate reviews reveals the need for standardization of license amendment application reviews and integration of technical review conclusions into licensing and licensing basis documents.

Another indication of the need for guidance regarding scope and depth of uprate reviews comes from the independent safety assessment of Maine Yankee. The assessment team identified mechanical components for which operability at the upgraded power level could not be confirmed.

The scope and depth of the uprate reviews varied substantially. This conclusion is apparent, not only from Table 3 and the above discussion of how codes were considered, but also from the identification in the SERs of the reviewers that contributed to the SERs. Some SERs did not list contributors; others identified only the project manager. Of those that listed technical reviewers, the number of technical review branches identified ranged from two to eight, and those that listed eight did not list the same eight.

The uprate applications and reviews surveyed covered some or all of Standard Review Plan (SRP) Sections 4.2, Fuel System Design; 4.3, Nuclear Design; 4.4, Thermal and Hydraulic Design (of core); and 15, Accident (and transient) Analysis; but several do not go beyond the scope of these sections. The scope of the later uprate reviews is more extensive than that of the earlier reviews, since applicants for the later uprates had the benefit of the GE and Westinghouse guidance for uprate submittals. However, the team found some variation in scope and some variation in the number of technical review branches identified, even among the later reviews. The significance of the variation in scope can be assessed by involving the technical review branches in the development of a standard review procedure for power uprates. (See Section 3.5, recommendation (2).)

Usually when a power uprate SER refers to a re-analysis, the SER doesn't state whether the re-analysis was done with the original code or with another code. In most of the cases where the licensee performed analyses for the first time using a new code, there is no evidence that an evaluation was performed to determine the effect on safety margin of using the new code. Future SERs should make clear whether the original code was used. The reviewer should identify all codes used in uprate analyses and should also identify the codes that have been approved for performing those analyses for that reactor. If a different code has been used, a baseline run at the old power level should be made with the new code. This baseline run will assist in estimating the effect on safety margin from the new code, as opposed to the effect of the uprate. The reviewer should assess whether approved codes are up to date with respect to relevant changes in knowledge, regulations, guidance, and changes in the plant, and should assess the use of approved codes with respect to code SER conditions. For unapproved codes, an assessment of the codes and their application to the analyses is needed.

Uprate reviews should consider all cumulative potential decreases in safety margin that have occurred, over the years (since the systems were last analyzed) from successive plant and procedure modifications, including the proposed uprate. The team did not find evidence that this was consistently done.

A recent situation at Brunswick suggests that reviewers should also compare the results of uprate analyses with the technical specifications to confirm that they are consistent. A 5% power uprate amendment was issued for Brunswick on November 1, 1996. During the course of the power uprate review the licensee and NRC staff believed that the torus design temperature listed in the UFSAR of 220F was correct. On November 4, 1996, the licensee reported that the Technical Specifications (Design Features section) listed a more restrictive value of 200F for the torus design temperature. At the former licensed power level of 2436 MWT, the licensee will not exceed the 200F limit under accident conditions; however, they could exceed it under power uprate conditions (2558 MWT) for the Station Blackout event. (The licensee has committed to limit power to the former licensed power level until this issue is resolved.)

Required changes resulting from the review and open items at the end of the review should be incorporated in license conditions or technical specifications. Again, it does not appear that this was consistently done.

The team found no indication that reviewers identified changes that should be made to the UFSAR as a result of the uprate.

Some of the above observations have implications for license amendments in general. The observations regarding variation in scope of reviews, number of technical review branches involved, apparent lack of review of selection and use of some codes, lack of consideration of cumulative potential decreases in safety margin, and lack of integration of uprate analyses and review conclusions into licensing and licensing basis documents have general applicability.

Some observations from other sources confirm the need for general guidance for reviewing license amendments. Several individual plant conversions from unique plant technical specifications to the standard technical specifications were done without technical branch involvement in the review or concurrence on the conversion packages. Among the conclusions in the Millstone Lessons Learned Task Group Report is that the NRC's licensing process should better identify the important aspects of plant-specific licensing actions and assure that those aspects are captured in the appropriate licensing document, and the inspection and licensing processes should work together more closely in choosing and verifying those aspects that need verification. In the recently concluded generic spent fuel pool action plan, the staff found that some plants' UFSARs did not reflect modifications that had been made to their spent fuel pools. In the same study, the staff also found that certain parameters identified in SERs were critical and should be, but were not, specified for followup in the inspection program. A recent event notification documented an instance in which an amended technical specification was inconsistent with its basis.

3.5 Recommendations

Based on the team's review, the following recommendations have been developed:

(1) **The PM Handbook should be changed to highlight the need to confirm that power uprates are incorporated into the next update of the UFSAR.** The project manager (PM) is not required to review UFSAR updates for incorporation of license amendments, such as power uprates. Section 3.4.13 of the PM Handbook states that UFSAR changes should be reviewed to ensure that they are covered by licensing actions, 10 CFR 50.59 reports, or regional inspection activities. However, it does not address verification that license amendments are appropriately reflected in UFSAR updates. Review guidance should be provided for the PM with respect to what information from a licensing application should be incorporated in the UFSAR.

(2) **A power uprate review procedure should be developed.**

An uprate review procedure is needed to guide project managers and technical reviewers. The staff has a Standard Review Plan, but that document guides the technical review of specific systems and parts of an application for a construction permit or operating license. The staff has no procedure for handling an application for a power uprate or other application for a license amendment. No guidance, such as an NRR office letter, tells the PM or technical reviewers what the scope of an uprate review should be and what questions to ask and points to consider.

The proposed procedure should: (1) specify how analytical codes and their use should be reviewed, (2) include guidance on review of licensee analyses and technical specifications, (3) alert the reviewer to the need to consider all cumulative potential decreases in safety margin that have occurred over the years (since the systems were last analyzed) from successive plant and procedure modifications, including the proposed uprate, (4) identify the technical review branches that should contribute to the review, (5) specify what information should be provided, by the PM, on the history and open items for the plant, (6) specify what followup is needed regarding license conditions and UFSAR update, and (7) clearly indicate the differences between a stretch power uprate review and an extended power uprate review. The scope and depth of review should be established based on a review of (1) the Standard Review Plan, as supplemented by the results of the Standard Review Plan Update and Development Program, (2) the staff position paper and the SER on GE's BWR power uprate generic analyses, and (3) uprate submittals that were based on the GE and Westinghouse topical reports on uprates.

In the NRR Task Action Plan for the BWR Extended Power Uprate Program, the staff is already planning to develop a standard review procedure for BWR uprates, based on (1) the staff position paper and the SER on GE's BWR extended power uprate generic analyses, and (2) the SER on the lead plant. A review procedure for PWRs is also needed.

(3) **The GE generic power uprate program and extended power uprate program should be reviewed against the standard review procedure that is developed for power uprates, and adjustments should be made to the programs as appropriate.**

(4) **The completed power uprates for which supplementary reviews were performed should be re-evaluated as appropriate against the criteria developed in the standard review procedure for power uprates, to confirm the general conclusions of the reviews.**

These re-evaluations need not be done immediately. All of the completed power uprate reviews were for uprates of approximately 5 percent or less. In a study documented in NUREG-1230, "Compendium of ECCS Research for Realistic LOCA Analysis," December 1988, the staff concluded that the potential risk impact from a 5 percent power increase is expected to be small and likely to be within the uncertainty of the present PRA methodology. Furthermore, in the Statement of Considerations for the 1988 rule revising the ECCS acceptance criteria, the Commission stated the position that the increase in fission products available for release during a postulated core meltdown, due to a 5 percent power increase, is negligible compared to the uncertainty in fission product release. Thus, these power uprates do not appear to pose an immediate safety issue.

(5) **The NRR office letter on license amendments, Office Letter 803, should be revised to provide additional general guidance for planning, performing, and closing out a review of a license amendment request.** Because a license amendment has potential health & safety implications, amendments should be planned and closed out carefully. **The technical branches that have responsibility for the technical areas involved in the amendment should be given the proposed amendment for review. Those technical branches involved in the amendment should develop a position, at the outset of the review, on what the scope & depth of a review of such an amendment request should be, if such a position has not already been developed. What documents, or portions of documents, should be reviewed or followed should be indicated. The review should be followed up with appropriate integration of analyses and review conclusions into licensing and licensing basis documents and the inspection program. For any staff SER that approves the use of a new code or methodology, it should be identified in the proposed Tech spec bases section. The PM should ensure that the appropriate parts of every license amendment, such as power uprates, are incorporated into the UFSAR. In addition, before concurring on the action, the PM's and technical reviewers' supervisors should verify that the review has been performed properly. This office letter, which can include the guidance on uprate reviews, should be incorporated into NRR's training program.**

4. ISSUE CLOSURE AND INTERFACE ISSUES

4.1 Closeout of TMI Action Items

The team contracted with Scientech, Inc., to conduct a document search to locate and compile the closeout documentation for TMI Action Plan Items II.K.3.5, II.K.3.30, and II.K.3.31 for all plants currently holding an operating license. Documents were examined to develop a general understanding of the review process applied to each item by the staff and to identify potential problem areas in the closeout. Specifically, technical staff involvement in the closeout of each item for each affected plant was assessed, along with management oversight of the closeout documents and safety evaluations in the closeouts.

4.1.1 CLOSEOUT ISSUES

4.1.1.1 TMI Action Plan Item II.K.3.5

This action plan item addressed trip criteria for reactor coolant pumps (RCPs) at PWRs during LOCAs. The NRC staff issued Generic Letters (GLs) 83-10a through 83-10f to PWR applicants and licensees providing guidance for the development of either satisfactory setpoints for RCP trip or technical bases for allowing continued RCP operation in the event of a small-break LOCA (SBLOCA).

In GL 85-12, the staff concluded that the generic information provided by the Westinghouse Owners Group (WOG) in support of alternative RCP trip criteria was acceptable on a generic basis. Applicants and licensees of Westinghouse plants were requested to choose from among three alternate RCP trip criteria.

In GL 86-05, the staff concluded that the generic information provided by the B&W Owners Group (BWOG) in support of the loss-of-subcooling RCP trip criterion was acceptable. The staff also requested plant specific information about instrument uncertainties, potential RCP problems, operator training, and procedures.

In GL 86-06, the staff concluded that the generic information provided by the Combustion Engineering Owners Group (CEOG) in support of the trip two/leave two staggered RCP trip criterion was acceptable. The staff also requested plant specific information about instrument uncertainties, potential RCP problems, operator training, and procedures.

The technical staff reviewed the plant-specific information and prepared SERs for about half of the affected plants between 1986 and 1988. Letters from PMs forwarded the SERs to the licensees and documented closeout of the issue. By memorandum to all Project Directors dated March 3, 1989, the Chief of the Reactor Systems Branch stated that the reviews of plant-specific information had progressed to a degree that the staff could conclude that the BWOG, CEOG, and WOG methodologies had significantly improved reactor safety and that there were no major safety-significant concerns for the plant-specific information. On this basis, the issue was considered closed for all remaining plants except Haddam Neck. From March 1989 through early 1990, the issue was closed for the remaining plants (except Haddam Neck) by letters from the PM to the licensee using the wording provided in the March 3, 1989, memo.

Haddam Neck remained open because the Connecticut Yankee Atomic Power Company did not endorse the WOG methodology. The issue was closed for Haddam Neck after staff review and approval of the plant-specific SBLOCA analysis, which indicated that RCP trip was not required for Haddam Neck. The closeout was documented in a letter from the PM to the licensee dated May 10, 1989, in which the technical staff and the Project Director concurred.

4.1.1.2 TMI Action Plan Items II.K.3.30 and II.K.3.31

In response to these items, GE provided information to demonstrate that its existing SBLOCA evaluation model (EM) complied with 10 CFR 50.46, taking into consideration comparisons with experimental data. By letter to GE dated December 13, 1983, the staff forwarded its SER on the GE submittal. The staff concluded that the GE EM was "very conservative" in predicting peak cladding temperatures (PCTs) for SBLOCAs and was therefore acceptable for prediction of SBLOCA transients in licensing submittals. In December 1983 and January 1984, letters to GE licensees from licensing branch chiefs (projects) in NRR, stated that, based on the generic GE SE, Items II.K.3.30 and II.K.3.31 were closed for those plants. Since the existing EM was demonstrated to be acceptable, no further plant-specific analysis was required.

Within the group of facilities receiving engineering services from Yankee Atomic Electric Company (the Yankee group), Vermont Yankee was covered by this generic effort. Items II.K.3.30 and II.K.3.31 were closed for Vermont Yankee by letter dated January 4, 1984.

Four plants under construction (Grand Gulf, La Salle, Susquehanna, and WNP-2) had committed to the GE generic effort. The staff accepted those commitments in the respective SERs. The team was unable to locate documentation of the closeout of this item. Supplemental SERs for each plant and the respective dockets were searched for the period immediately following issuance of the generic SER on the GE EM. It is recommended that the applicability of the generic GE EM to these four facilities be verified. (See Section 4.1.3.)

Westinghouse submitted a new licensing EM, NOTRUMP, as described in Westinghouse topical report, WCAP-10079. By letter to Westinghouse dated May 21, 1985, the staff forwarded its SER regarding NOTRUMP. The staff concluded that NOTRUMP was an acceptable computer program for use in performing licensing calculations of SBLOCAs for Westinghouse-designed NSSSs. The staff also stated that its review fulfilled the requirements in TMI Action Plan Item II.K.3.30 for computer code validation, and that NOTRUMP was therefore the acceptable code for TMI Action Plan Item II.K.3.31 application. In May through July 1985, affected licensees were informed in letters from NRR licensing branch chiefs that, based on the May 21, 1985, SER, Item II.K.3.30 was closed for their plants and that plant-specific applications of NOTRUMP should be submitted within one year. On June 11, 1986, Westinghouse submitted topical report WCAP-11145, "Westinghouse Small Break LOCA ECCS Evaluation Model Generic Study with the NOTRUMP Code." In an SER dated October 6, 1986, the staff found that in WCAP-11145, the guidance of TMI Action Plan Item II.K.3.31, as clarified by GL 83-35, had been satisfied. Subsequent letters from project managers to licensees informed them that, based on their referencing WCAP-11145, Item II.K.3.31 was closed. Therefore all Westinghouse plants, except Haddam Neck, had approved SBLOCA methodologies using the approved WCAP-11145.

Within the Yankee group, Seabrook was covered by the Westinghouse generic effort. The staff closed Item II.K.3.30 for Seabrook by letter dated June 11, 1985, forwarding the staff's generic SER of May 21, 1985. By letter dated January 6, 1987, the staff accepted the licensee's referencing WCAP-11145 for Seabrook and closed Item II.K.3.31.

In response to TMI Action Plan Item II.K.3.30, the CEOG elected to justify continued acceptance of the CEFLASH-4AS computer program for SBLOCA evaluation. The CEOG response was transmitted in a generic topical report CEN-203, "Response to NRC Action Plan Item II.K.3.30 Justification of Small Break LOCA Methods." By letter to the CEOG dated June 20, 1985, the staff forwarded its SER regarding CEFLASH-4AS. The staff concluded that, pending an acceptable benchmark comparison of CEFLASH-4AS with Semiscale experiment S-UT-08, Action Item II.K.3.30 was resolved for CE plants. With an acceptable calculation/data comparison, the staff found that the requirements to perform a plant-specific analysis would no longer apply to the CE plants except Maine Yankee. On December 31, 1985, the CEOG submitted topical report CEN-203, Revision 1-P, Supplement 3 and Revision 1-P, Supplement 4 in response to the staff's concern that the CEFLASH-4AS computer program might not be able to calculate the initial rapid drop in water level experienced in the simulated reactor vessel in the Semiscale Test S-UT-08. The staff provided its review of these supplements in a letter dated February 11, 1987. The staff found that the CEFLASH-4AS program acceptably calculated the test. On this basis, Action Plan Items II.K.3.30 and II.K.3.31 were closed for the CEOG licensees except MYAPCo (Maine Yankee). The team's review raised a question regarding the applicability of this generic effort to San Onofre Nuclear Generating Station, Units 2 and 3 (SONGS 2/3) which is addressed in Sections 4.1.1.3 and 4.1.3.

In response to TMI Action Plan Item II.K.3.30, the BWOG submitted a modified CRAFT2 EM in topical report BAW-10092(P), Rev. 3, "CRAFT2 - FORTRAN Program for Digital Simulation of a Multinode Reactor Plant During Loss of Coolant" (Proprietary) and non-proprietary version designated BAW-10154(NP). By letter to Babcock & Wilcox dated June 27, 1985, the staff issued a generic safety evaluation accepting the B&W topical reports for referencing in license applications. In letters to the affected licensees in July and August 1985, the staff requested that each licensee confirm within 45 days that the BWOG position on noncondensable gases was the licensee's position and that the BWOG commitment to compare the CRAFT2 LOCA analysis results with the Multiloop Integral System Test Facility (MIST) results was also the licensee's commitment. The team verified receipt of each licensee's confirmatory response. The BWOG provided the MIST LOCA test results on July 31, 1989. In accordance with the guidance regarding Item II.K.3.31 in GL 83-35, the BWOG submitted topical report BAW-1976 to demonstrate that the generic CRAFT2 analysis results conservatively bounded the existing licensing basis ECCS SBLOCA analyses for lowered loop B&W plants. The BWOG also submitted topical report BAW-1981 to provide the bounding analysis for raised loop plants. The staff reviewed these topical reports and closed II.K.3.31 for the affected plants in mid-1989. There were no B&W facilities not covered by this effort.

4.1.1.3 Technical Staff Involvement

The closeout documents were reviewed to assess technical staff involvement. Technical staff involvement included preparation of a generic SER, preparation of a plant-specific SER, or concurrence in a closeout letter. Where the closeout was documented in a Supplement to the Facility Safety Evaluation Report, technical staff involvement was inferred. In some cases, internal memoranda were located to verify technical staff involvement.

One concern was identified. By letter dated May 21, 1987, Southern California Edison Company cited the staff's approval of the CEOG generic methodology as the basis for concluding that no further licensee action was required regarding TMI Action Plan Items II.K.3.30 and II.K.3.31 for San Onofre Nuclear Generating Station (SONGS), Units 2 and 3. By letter dated July 17, 1987, the PM informed the licensee that the staff considered those items closed for SONGS 2/3. The PM cited as basis the licensee's letter of May 21, 1987. The PM's letter did not have concurrence from the technical staff. The team was unable to locate an internal memorandum providing input to the PM's letter. The project director (PD) concurred in the letter. (See Section 4.1.3.)

4.1.1.4 Management Oversight

The team examined each closeout document to assess management oversight and involvement. Where the closeout was documented in a Supplement to the Facility Safety Evaluation Report, management oversight was inferred. Concurrence by an acting Project Director other than the originator of the document was considered to constitute acceptable management oversight. Lack of management oversight was deemed to exist when no management concurrence was indicated or when the originator of the document concurred on behalf of management.

The team identified three closeout letters which were issued without Associate Director for Projects (ADPR) management concurrence. Each of these letters enclosed a safety evaluation prepared by the technical staff regarding plant-specific analyses pursuant to Item II.K.3.31. These closeout documents were similar to several dozen others which were issued with management concurrence. Issuance of these documents did not raise any concerns.

The team identified nine closeout documents in which the originating PM concurred on behalf of management, signifying acknowledgement that management involvement was required. One of the nine was the May 8, 1989, letter which closed Item II.K.3.31 for Maine Yankee. Two others forwarded safety evaluations prepared by the technical staff for plant-specific analyses pursuant to Item II.K.3.31. These two documents were similar to several dozen others which were issued with management concurrence. Five of the nine letters closed Item II.K.3.5. Two of the five issued safety evaluations which had been prepared by the technical staff and were similar to other closeout letters which included management concurrence. The team compared the remaining three to the generic closeout guidance provided by the technical staff in a memorandum dated March 3, 1989. The three letters were consistent with the technical staff closeout guidance and similar to several dozen other letters issued with appropriate management concurrence. The last item in which the originating PM concurred on behalf of management was a TS amendment regarding a change in fuel supplier. As part of the review of that amendment, the technical staff had reviewed the licensee's plant-specific SBLOCA analysis using the generic Westinghouse EM which had previously been approved for the subject plant pursuant to Item II.K.3.30. In conjunction with the issuance of the amendment, the staff closed Item II.K.3.31. The team had no concerns about this item. Of the nine documents, the team found that eight were acceptable as they contained appropriate technical justification and technical staff involvement. One product, the May 8, 1989, letter to Maine Yankee, was found to be an outlier.

In summary, the team identified three documents which had no management concurrence, and nine documents in which the PM concurred on behalf of management. The team compared these documents to other closeout documents which had appropriate management concurrence. The May 8, 1989, letter to Maine Yankee, was identified to be the sole outlier.

4.1.1.5 Safety Evaluations

The closeout of each item for each affected plant was reviewed for documentation of the staff's evaluation of the licensee's response. Item II.K.3.30 was typically closed via a generic SER forwarded to the affected owners group plants. Item II.K.3.31 was also typically closed by a staff SER describing the staff's review of the plant-specific or generic bounding SBLOCA analysis. Notable exceptions include the Maine Yankee application regarding RELAP5YA and the Haddam Neck application regarding NULAP5.

The Maine Yankee code RELAP5YA was approved in a safety evaluation dated October 14, 1988, which was forwarded to the licensee by letter dated January 17, 1989. The Maine Yankee SER contained 12 conditions on the acceptable application of the code. Maine Yankee was the only plant that did not make a submittal to satisfy the guidance in Item II.K.3.31.

The Haddam Neck code, NULAP5, and a plant-specific analysis using NULAP5 were approved with nine conditions in a safety evaluation dated August 3, 1988. The licensee submitted another plant-specific LOCA analysis in January 1991 using a best estimate upper plenum injection model in conjunction with an amendment application related to conversion from stainless steel clad fuel to Zircaloy clad fuel. The staff accepted that analysis in conjunction with the issuance of the fuel conversion amendment on January 17, 1992. The acceptance included exemptions for the model from certain requirements of Appendix K to 10 CFR Part 50.

Item II.K.3.5 was closed for about half of the affected plants (PWRs) before 1989 by issuance of SERs describing the staff's evaluation of the plant-specific information provided in response to GLs 85-12, 86-05 and 86-06. By memorandum dated March 3, 1989, the Reactor Systems Branch provided generic closeout guidance for the remaining plants (except Haddam Neck) stating that the staff's review of the plant-specific information had concluded that no safety-significant concerns remained. No SERs were issued with these closeout letters. The Haddam Neck closeout is discussed under Item II.K.3.5 above and enclosed a safety evaluation prepared by the technical staff.

For Maine Yankee, Item II.K.3.5 was closed via SER dated April 15, 1986. The SER was prepared by the technical staff after meeting with the licensee and receiving the licensee's responses to staff questions on non-LOCA issues related to Item II.K.3.5. The licensing branch chief concurred in the issuance of the SER to the licensee.

4.1.2 FOLLOWUP/VERIFICATION PROCESS

The Safety Issues Management System (SIMS) indicates that the three subject TMI Action Plan Items did not require verification. Nonetheless, NUDOCS was used to identify inspection reports related to these three items. Four reports for II.K.3.30 and another four related to II.K.3.31 were identified. These Region inspection reports documented NRR closure of the cited issue but did not indicate any further follow up. The team notes that the review of the plant-specific analysis submitted pursuant to II.K.3.31 using the EM approved by the staff pursuant to II.K.3.30 provided verification of licensee implementation of the approved EM.

The team identified 48 inspection reports related to Item II.K.3.5. A representative sample of these reports, including reports from each Region, were evaluated. The inspections examined licensee responses to the GLs related to Action Plan Item II.K.3.5 and verified that the statements made by the licensees in their submittals had been appropriately incorporated into plant procedures. These reports generally concluded that licensees had conformed to their commitments regarding reactor coolant pump trip. Where concerns were raised by the inspectors, appropriate inspector follow items were opened. The team concluded that the verification performed regarding Item II.K.3.5 provided reasonable assurance that licensees had conformed to their commitments regarding implementation of procedures regarding reactor coolant pump trip.

4.1.3 RECOMMENDATIONS

Based on the team's review, the following recommendations have been developed:

(1) **The technical staff should verify the applicability of the GE effort to Grand Gulf, La Salle, Susquehanna, and WNP-2, and document the basis in a memo to the docket file.** The team was unable to verify the closeout of TMI Action Plan Items II.K.3.30 and II.K.3.31 for Grand Gulf, La Salle, Susquehanna, and WNP-2. Each of these facilities' licensees had committed to the generic GE effort which the staff found to be conservative. Therefore the team did not have an immediate safety concern.

(2) **The technical staff should verify the applicability of the CEOG effort to SONGS 2/3 and document it in a memo to the docket file. (See Sections 4.1.1.2 and 4.1.1.3.)** Items II.K.3.30 and II.K.3.31 were closed for SONGS 2/3 based on a letter from the licensee stating that the staff's generic approval of the CEOG methodology applied to that facility. No technical staff involvement in that closeout was documented. The team did not have an immediate safety concern because the CEOG effort involved demonstration that the existing EM was adequate taking into consideration comparison to test facility experiment results. Provided that the CEOG methodology is applicable to SONGS 2/3, no concern exists.

(3) **A Temporary Instruction should be developed to enable inspectors and/or project managers to audit the implementation of Item II.K.3.31 by reviewing plant operating parameters derived from the cycle-specific analysis to understand that a logical progression of safety analysis has been followed. Specifically, selected plant parameters should be validated against the SBLOCA code of record, and the code of record should be assessed against the original approved SBLOCA code developed to conform with the guidance in Item II.K.3.30. This entire process should be completed as part of a 10 CFR 50.59 review. (See Section 4.1.2)** Region inspectors conducted selected reviews of the implementation of reactor coolant pump trip strategies. This adequately addressed implementation of licensee commitments regarding Item II.K.3.5. Regarding II.K.3.30 and II.K.3.31, no inspection activity was identified which verified licensee implementation of the ECCS EMs following closeout of Items II.K.3.30 and II.K.3.31. The Maine Yankee experience involves a licensee making apparently unacceptable changes to the approved model without reporting to the Commission as required by 10 CFR 50.46.

4.2 Staff Interfaces

The team assessed various staff guidance documents, including Management Directives, NRR Office Letters, and the Project Manager's Handbook (NUREG/BR-0073, Revision 1), for guidance regarding documentation of communications with licensees, technical staff involvement, signature authority, and tracking of licensee commitments.

In addition to evaluating the adequacy of staff guidance documents, current staff performance in the areas discussed below was benchmarked by reviewing selected documents generated by the projects organization within NRR during the first half of calendar year (CY) 1996. The documents reviewed included a broad sampling of licensing action approvals and denials, and generic issue evaluations. The sample included at least two documents for each operating power reactor. A total of 257 documents were reviewed. The sample included 115 amendments, of which 5 were emergency amendments and 7 were exigent amendments. The remaining documents included exemptions, reliefs, Code cases, task interface agreements, and plant-specific topical report evaluations. The results of the review are indicated in the appropriate sections below.

4.2.1 DOCUMENTATION OF COMMUNICATIONS WITH LICENSEES

The team found little written guidance to project managers on documentation of communications with licensees. Discussions with project managers indicated that they are aware of the need to document communications with licensees when those communications form or contribute to the basis for a licensing decision. Some PMs maintain contemporaneous notes of phone calls. Most PMs request that licensees provide information on the docket which was provided in telephone conversations when that information forms the basis of a licensing decision. On rare occasions, telephone conversations are documented in a memo to the docket file. These memos were not used to form the basis for a licensing decision.

A NUDOCS search for documentation of telephone conversations during the first half of CY 1996 revealed thirteen documents. Only three of those involved the licensee. It was concluded that PMs generally do not document telephone conversations that are deemed to be of minor significance, and that for telephone conversations involving information of significance to a licensing decision the licensee documents the information in a letter on the docket to the staff.

It is recommended that NRR management clearly define and document expectations of project managers regarding documentation of communications with licensees. These expectations should take into consideration that PMs routinely conduct mundane conversations for which formal documentation would constitute an undue burden. Any information which forms the basis of a pending staff licensing decision must be provided by the licensee on the docket. (See Section 4.2.5.) Of note is that the Process Improvement Plan of ADPR includes development of a position in this area.

4.2.2 TECHNICAL STAFF COORDINATION AND CONCURRENCES

The team evaluated various guidance documents related to coordination between the project manager and the technical staff and to appropriate levels of concurrence. These guidance documents included NRR Office Letters, the Project Manager's Handbook, and NRC Management Directives.

NRR Office Letter (OL) 803, Revision 1, "Technical Specifications Review Procedures," dated February 27, 1996, provides guidance and assigns responsibilities to NRR staff involved in the review of license amendment requests or the development of new technical specifications. OL 803 indicates that the project manager is responsible for determining the appropriate level of technical staff involvement in the review of a proposed license amendment. Specifically, the OL states that "Staff from the technical branches shall work with the project manager, *as requested* ...," and "*If asked by* *the project manager*, personnel from the technical branches shall assist ..." (emphasis added). The OL goes on to state that the project manager shall coordinate technical staff involvement in the review. Finally the OL indicates that the project manager shall ensure that the review and concurrence chain includes all of the individuals responsible for the quality of the document. The technical staff should review and concur if (1) the technical staff did not prepare the SER, or (2) the technical staff prepared the SER and the project manager has made substantial changes to the SER.

OL 803 provides guidance and assignment of responsibility to ensure appropriate coordination between the project manager and the technical staff, and appropriate technical staff concurrence in the review of license amendment applications. (But see recommendations 3.5(2) and 3.5(5).) OL 803 does not address coordination of other licensing actions such as requests for reliefs or exemptions. It also does not address coordination of other tasks such as resolution of generic issues, event reports, 50.59 reviews, response to task interface agreements from the regions, requests for interpretations from licensees, 2.206 Director's Decisions, and controlled correspondence.

Of the 257 documents reviewed, 32 did not contain direct technical staff involvement by either preparation of or concurrence in the document. The team reviewed the content of these documents. Typically they (1) corrected typographical errors in previous NRC documents, (2) involved administrative changes to plant technical specifications, (3) enclosed standard safety evaluations for Generic Letter responses, or (4) denied licensee applications. The team did not identify any concerns regarding the content of these 32 documents.

In evaluating the ability of the PM to fulfill expectations regarding obtaining appropriate concurrences, it was noted that the PM must have clear guidance on which branch(es) in the ADT organization are to be involved in each individual review. To ensure that the PM has adequate information on organizational responsibilities within NRR, function-oriented organizational charts reflecting those responsibilities at the Section level should be promulgated with each re-organization. (See Section 4.2.5. Also see recommendation 3.5(2).)

4.2.3 SIGNATURE AUTHORITY

The staff guidance documents related to signature authority and concurrence were evaluated. These guidance documents included NRR Office Letters, the Project Manager's Handbook, and NRC Management Directives.

On review of the May 8, 1989, letter from the project manager to Maine Yankee, it is the team's understanding that the licensee was of the opinion that this letter superseded a previous staff position that required the submittal of a plant-specific SBLOCA analysis using the previously approved code, RELAP5YA. The staff's signature authority, as designated in OL 101, "Delegation of Signature Authority," does not specifically address the circumstance of general correspondence to a licensee which reverses an existing staff position. However, management expectations and the intent of OL 101 suggests that the May 8, 1989, letter should have required at least the technical staff Division Director's signature, possibly the Office Director's. In addition, the change of licensee commitments previously made in writing by a telephone conversation in which the commitment revisions involve substantive regulatory and technical issues clearly should have involved senior technical staff management involvement and concurrence.

Of the 257 documents reviewed, in ten cases the originating project manager signed or concurred as the acting project director, effectively bypassing management oversight. OL 101, Revision 11, issued on August 9, 1996, includes a footnote which states that, "To ensure appropriate independent review, when a non-supervisory staff member is acting for the supervisor, the staff member should not concur and/or sign on their own work. The staff member should obtain concurrence or signature from another cognizant supervisor, or the next higher level of management." The team concludes that this revision addresses the concerns identified from the May 8, 1989, letter and suffices as direction to ensure appropriate management oversight in the future where supervisor concurrence is required.

Within the ADPR organization, PDs are afforded latitude in the amount of involvement required in overseeing products issued from NRR. Specifically, this effort identified that supervisor concurrence is not always required. Some senior, experienced project managers are authorized to sign and issue various correspondence without obtaining PD concurrence. Of the 257 documents reviewed, 46 did not include any PD concurrence. Of those 46, the project director was not on distribution for 12. Thus about 18% of the products reviewed had no opportunity for PD oversight.

OL 101 indicates that the Project Director signs approvals and denials specified in 10 CFR 50.55a. Of the 257 documents reviewed, 66 involved approvals or denials of inservice inspection and inservice testing (ISI/IST) programs, Code reliefs, and Code alternatives. Of the 66, eight were inappropriately signed by the project manager. The project director concurred in five of those eight. The PD was on distribution of the other three. The team did not have any concerns regarding the content of these eight documents. No signature authority problems were identified for any other type of document.

It is recommended that ADPR staff be reminded of the signature authority for approval or denial of licensing requests pursuant to 10 CFR 50.55a. (See Section 4.2.5.)

An inconsistent standard has evolved within the ADPR organization. In some cases, if a Senior Project Manger concurs as acting PD in an amendment package he/she originated, such action would be contrary to Revision 11 to OL 101. In other cases, some project managers are not even required to obtain PD concurrence in an amendment. In the case of the May 8, 1989, letter to Maine Yankee, the concurrence by the Project Manager in lieu of his supervisor was cited by the OIG as a significant weakness.

It is recommended that management ensure uniformity and consistency in assuring appropriate oversight of the licensing program. Specifically, license amendments are significant regulatory actions requiring, as a minimum, first-level supervisor oversight. It is noted that such actions contain other checks and balances in the process, including independent reviews by the licensing assistant, technical staff, and the Office of General Counsel. In addition, other licensing actions such as reliefs and exemptions require a higher level of management involvement. Products that would not normally require Senior Executive Service (SES) level involvement unless a sensitive or unusual issue was involved include meeting summaries, memoranda to file, and routine correspondence. This matter could best be addressed in a further revision of OL 101. (See Section 4.2.5.)

4.2.4 TRACKING LICENSEE COMMITMENTS

Staff guidance related to licensee commitments contained in the Project Manager's Handbook was evaluated.

The guidance discusses written utility commitments and notes that such commitments are not binding on the utility. It further states that, to the extent that such commitments are judged to be a necessary element to support an acceptable finding, they should be spelled out clearly in the SER. The inference drawn is that, although the commitment is not binding on the utility, a change to the commitment would undermine the basis for the staff's acceptance and therefore requires prior staff approval. The guidance notes that a commitment may be given "enforcement strength" by order. The guidance states that "any failure to live up to [a commitment imposed by order] bears a stronger enforcement penalty than if it were simply a written commitment." From this a reader could infer that there was some binding enforcement sanction, albeit a lesser one, for failure to live up to a written commitment without a confirmatory order. This contradicts the aforementioned statement that such a commitment is not binding on the utility.

The guidance addresses the handling in the licensing process of written utility commitments which are deemed to be a necessary part of approval of a licensee's proposal. The guidance does not address tracking or follow-up verification of such commitments. It also does not address the management of lower order commitments such as schedular commitments, or commitments provided verbally (e.g., at inspection exit meetings, licensing meetings, or in telephone conversations.)

Of the 257 documents reviewed, 13 identified licensee commitments which formed the basis of the staff's approval and 6 placed conditions on the staff's approval.

In December 1995, the Commission endorsed the Nuclear Energy Institute (NEI) guidance on licensee commitment tracking (SECY-95-300). In a staff meeting in early 1996, project managers were informed that licensees may be implementing the guidance and that project managers would be expected to review their licensee's commitment tracking system. No formal guidance has been issued.

It is recommended that NRR management issue clear, formal guidance on the various levels of licensee commitments, including expectations of the project managers regarding commitment tracking and verification, interaction with the Regions in verification of licensee commitments, and project manager oversight of licensee commitment tracking systems. (See Section 4.2.5.) The team notes that ADPR's Process Improvement Plan includes development of guidance on documentation of verbal commitments.

4.2.5 RECOMMENDATIONS

(1) **NRR management should clearly define and document expectations of project managers regarding documentation of communications with licensees.** These expectations should take into consideration that PMs routinely conduct mundane conversations for which formal documentation would constitute an undue burden. They should also recognize that the staff has routinely relied on licensee recollections of past conversations because no NRC documentation of the conversations survived. Any information which forms the basis of a pending staff licensing decision must be provided by the licensee on the docket.

(2) **In evaluating the ability of the PM to fulfill the expectations regarding obtaining appropriate concurrences, it is noted that the PM needs clear guidance on which branch(es) in the technical staff are cognizant for each of the various technical areas. To ensure that the PM has adequate information on organizational responsibilities within NRR, function-oriented organizational charts reflecting those responsibilities at the Section level should be promulgated with each re-organization.**

(3) **ADPR staff should be reminded of the OL 101 signature authority for approvals and denials pursuant to 10 CFR 50.55a.** The team found that 12% of these items were signed by the project manager rather than the project director as required by OL 101. The project director had concurred in most of those documents.

(4) **NRR management should ensure uniformity and consistency in assuring appropriate oversight of the licensing program.** Specifically, license amendments are significant regulatory actions requiring, as a minimum, first-level supervisor oversight. The team notes that such actions contain other checks and balances in the process, including independent reviews by the licensing assistant, technical staff, and the Office of General Counsel. In addition, other licensing actions such as reliefs and exemptions require a higher level of management involvement. Products that would not normally require SES level involvement unless a sensitive or unusual issue was involved include meeting summaries, memoranda to file, and routine correspondence. This matter could best be addressed in a further revision of OL 101. The team notes that ADPR's Process Improvement Plan includes revision of OL 101.

(5) **NRR management should issue clear, formal guidance on the various levels of licensee commitments, including expectations for the project managers regarding commitment tracking and verification, interaction with the Regions in verification of licensee commitments, and project manager oversight of licensee commitment tracking systems.**

LIST OF ABBREVIATIONS

ABB	Asea Brown Boveri
ACRS	Advisory Committee on Reactor Safeguards
ADPR	Associate Director for Projects, NRR
ADS	automatic depressurization system
ADT	Associate Director for Technical Review, NRR
AFW	auxiliary feedwater system

ATWS	anticipated transient without scram
B&W	Babcock and Wilcox
BWR	boiling water reactor
BWOG	B&W Owners Group
CCW	component cooling water system
CE	Combustion Engineering
CEOG	Combustion Engineering Owners Group
COLR	core operating limits report
CRD	control rod drive
CY	calendar year
DBA	design basis accident
DRPE	Division of Reactor Projects I/II, NRR
DRPW	Division of Reactor Projects III/IV, NRR
DSSA	Division of Safety Systems and Analysis, NRR
ECCS	emergency core cooling system
EDO	Executive Director for Operations
EM	evaluation model
EQ	equipment qualification
FSAR	Final Safety Assessment Report
FTI	Framatome Technologies, Inc.
GDC	General Design Criterion
GE	General Electric
GL	generic letter
HVAC	heating, ventilating, and air conditioning
I&C	instrumentation and control
IEEE	Institute of Electrical and Electronics Engineers
ISI/IST	inservice inspection and inservice testing
LBLOCA	large-break loss-of-coolant accident
LCO	limiting condition for operation
LOCA	loss-of-coolant accident
MIST	Multiloop Integral System Test Facility
MOV	motor operated valve
MSLB	main steam line break
MWt	megawatts thermal
MYAPCo	Maine Yankee Atomic Power Company
NEI	Nuclear Energy Institute
NPSH	net positive suction head
NSSS	nuclear steam supply system
NRR	Office of Nuclear Reactor Regulation
OGC	Office of General Counsel
OIG	Office of the Inspector General
OL	NRR Office Letter
OTSG	once through steam generator
PCT	peak clad temperature
PD	project director
PDI-3	Project Directorate I-3, NRR

PM	project manager
PWR	pressurized water reactor
QA	quality assurance
RAI	request for additional information
RCP	reactor coolant pump
RCS	reactor coolant system
RCIC	reactor core isolation cooling system
RHR	residual heat removal system
RSG	recirculating steam generator
SASG	Analytical Support Group, NRR
SBLOCA	small-break loss-of-coolant accident
SBO	station blackout
SER	safety evaluation report, whether it is entitled "Safety Evaluation Report" or "Safety Evaluation"
SES	Senior Executive Service
SFP	spent fuel pool
SG	steam generator
SIMS	Safety Issues Management System
SLC	standby liquid control system
SONGS	San Onofre Nuclear Generating Station
SPC	Siemens Power Corporation
SRP	Standard Review Plan
SRXB	Reactor Systems Branch, NRR
TER	technical evaluation report
TMI	Three Mile Island
TS	technical specification
UFSAR	updated final safety analysis report
W	Westinghouse
WNP-2	Washington Public Power Supply System Nuclear Project No. 2
WOG	Westinghouse Owners Group
YAEC	Yankee Atomic Electric Company

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1. In response to the allegations received in December 1995, the NRC staff conducted an audit of the Maine Yankee SBLOCA and containment peak-pressure analyses at the licensee's engineering services contractor. As a result of this review, the staff issued an Order and Demand for Information on January 3, 1996, restricting power to 2440 MWt and containment pressure to 2 psig.