RESPONSE TO THE

NUCLEAR REGULATORY COMMISSION'S

DEMAND FOR INFORMATION

(NRC OI REPORT NO. 1-95-050)

by

Duke Engineering & Services, Inc.
Charlotte, North Carolina

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Executive Summary

On December 1, 1995, the Union of Concerned Scientists (UCS) sent a letter to the State of Maine Nuclear Safety Advisor stating that they had received anonymous documentation alleging that the management at Maine Yankee deliberately falsified reports to the NRC (Nuclear Regulatory Commission). NRC received these allegations on December 4, 1995 and sent an assessment team to Yankee Atomic Electric Company (YAEC) to investigate the allegations. On January 3, 1996 NRC issued a Confirmatory Order limiting power operation and containment pressure at Maine Yankee and a Demand for Information to Maine Yankee Atomic Power Company (MYAPCo). The order was based in part on NRC allegations that Maine Yankee could not demonstrate that the RELAP5YA computer code would reliability calculate the peak cladding temperature (PCT) for all Small Break Loss-of-Coolant Accidents (SBLOCAs) for Maine Yankee).

As a result of these allegations, YAEC and MYAPCo chartered three teams to look into these issues. YAEC and MYAPCo management formed a Response Team, composed of managers and technical specialists with responsibilities for the analyses in question, and an Independent Review Team, composed of managers and technical specialists with no prior responsibilities for the analyses in question, to investigate the allegations. In February, 1996, MYAPCo and YAEC executive management chartered an Assessment Team to determine the underlying cause(s) of the allegations. YAEC subsequently developed a schedule for and implemented corrective actions in a wide range of areas including organizational changes to clarify the relationship between Maine Yankee and YAEC, new engineering procedures, revisions to the Employees Concerns Program, revised training programs, revised commitment tracking, and a new Condition Reporting system.

On December 19, 1997, the NRC sent a letter to Yankee Atomic Electric Company and Duke Engineering and Services, Incorporated (DE&S) which formally transmitted a "Demand for Information to Yankee Atomic Electric Company and to Duke Engineering & Services - Re: Providing Inadequate Engineering Analyses and Materially Incomplete and Inaccurate Information to an NRC Licensee (NRC OI Report No. 1-95-040)." As a result of the issues described within the Demand, the NRC requested YAEC and/or DE&S to provide the following information:

A. "An explanation why, in the view of the matters set forth above [in the Demand], the NRC should permit any NRC Licensee to use the services of YAEC LOCA Group and/or DE&S, to the extent that YAEC LOCA Group was transferred to DE&S, to perform LOCA analyses or any safety-related analyses to meet NRC requirements."

B. "An explanation why the NRC should not consider the inadequate analyses, which apparently caused MYAPCo [Maine Yankee Atomic Power Company] to be in violation of NRC requirements, to be the result of willfulness, whether deliberateness or careless disregard, on the part of YAEC and/or DE&S personnel."

Also, in December, 1997 DE&S announced the acquisition of the Nuclear Services Division of YAEC. DE&S was aware of the situation between YAEC and NRC concerning Maine Yankee prior to this acquisition. With the issuance of the NRC Demand for Information (Demand) ongoing reviews and assessments were accelerated and several additional assessments were chartered in order to provide the requested response to NRC. The scope of these DE&S reviews and assessments was:

1. To independently review the findings, recommendations, and corrective actions taken from the three teams previously formed by YAEC,

2. To review the engineering and technical work processes and Quality Assurance programs at YAEC,

3. To independently review the technical issues concerning SBLOCA analyses

4. To determine, from a legal perspective, whether any of the personnel actions taken or decisions made related to the SBLOCA analyses were the result of willfulness,

5. To review a sampling of analyses performed by YAEC for other NRC licensees, specifically, Vermont Yankee and Seabrook.

The results of these reviews and assessments are summarized in the main body of this response. Copies of the various reports are provided as Appendices to this response. Ultimately, the information from these reports provided the necessary information to respond to the two specific items in Section V of the Demand. Additionally, in Section IV of the Demand, NRC raised "serious questions" in four areas to which DE&S provided responses.

Section III of the Demand described four specific technical issues concerning information supplied by YAEC to MYAPCo which was related to six apparent violations received by MYAPCo. DE&S addressed these technical issues specifically as part of this response since they dealt with the development and use of RELAP5YA SBLOCA computer code by YAEC which goes to the heart of the Demand.

The DE&S response to each of these items is briefly summarized below.

Demand (Section V, Item A):

> *An explanation why, in view of the matters set forth [in the demand], the NRC should permit any NRC licensee to use the services of YAEC LOCA Group and or DE&S, to the extent that YAEC LOCA Group was transferred to DE&S, to perform LOCA analyses or any safety-related analyses to meet NRC requirements*

Response:

> DE&S has found, based on its assessments, that performance of, and the products and services provided by, the acquired YAEC organizations are acceptable (e.g., compliant with NRC requirements). Meaningful improvements have been made during the last two years in the quality of YAEC procedures and the emphasis these procedures place on NRC requirements and NRC reporting. The technical quality of the work products being produced, as well as the professionalism and technical competence of the workforce, are consistently high. DE&S is systematically transitioning the acquired YAEC organizations to the DE&S project planning and work execution programs and procedures. The DE&S project planning process emphasizes clear definition and documentation of project-specific requirements, both technical and organizational, and training of project personnel on these requirements. DE&S has strengthened the Bolton office leadership with a proven nuclear industry executive, William H. Rasin. DE&S is also currently

jointly engaged with each of the nuclear clients formerly supported by YAEC to ensure that roles, responsibilities, and expectations are mutually understood and documented. For these reasons, and the nuclear commitment and experience presented by DE&S, the NRC should have a high degree of confidence that safety-related analyses, products and services provided by DE&S to NRC power reactor licensees comply with the principles of: (1) adherence to NRC requirements, (2) effective management control of safety-related activities, (3) accurate and complete communication with licensees and the NRC, and (4) conducting work in accordance with NRC requirements.

Demand (Section V, Item B):

An explanation why the NRC should not consider the inadequate analyses, which apparently caused MYAPCo to be in violation of NRC requirements, to be the result of willfulness, either deliberateness or careless disregard, on the part of YAEC and /or DE&S personnel.

Response:

DE&S has found, throughout its assessment process, that employees of the acquired YAEC organizations display a high degree of professionalism and technical competence. All individuals are open, honest, and communicative. While in certain instances, there may have been inadequate analysis, there was neither deliberateness nor careless disregard resulting from the deficiencies described in the Demand. DE&S found no willfulness on the part of the two individuals mentioned in the Demand and believes that these individuals are capable of conducting their activities in conformance with NRC requirements. DE&S also found no evidence that would question the sincerity and dedication of the acquired YAEC work force, or would otherwise prevent DE&S activities from being conducted in full compliance with NRC requirements.

Demand (Section III, Item A):

It was not possible to confirm that the limiting break had been identified and that the emergency core cooling system was capable of mitigating the most severe postulated break for Maine Yankee Cycles 14 and 15.

Response:

DE&S agrees that standard industry practice, as conducted by experts in LOCA analyses, is to utilize a code or set of codes with the capability to analyze all points within the prescribed break spectrum. As described within the Demand, the codes utilized by YAEC (i.e., RELAP5YA and WREM) did not have this demonstrated capability. YAEC had taken the position that analyses combined with an understanding of the physical phenomena occurring throughout the break spectrum provided the basis for compliance with the technical requirements of 10CFR50.46. DE&S believes that the reading of 10CFR50.46 by YAEC was understandable. DE&S also notes that the results of Maine Yankee SBLOCA analyses performed by YAEC are similar to results of SBLOCA analyses performed by other organizations. Thus, it is possible that YAEC's understanding of 10CFR50.46 may have received NRC approval had the Maine Yankee SBLOCA analyses and supporting documentation been submitted to the NRC for review. The error in judgement was not in the interpretation of the technical requirements of 10CFR50.46, but in the

manner in which compliance with 10CFR50.46 was demonstrated, or assumed to have been demonstrated. DE&S also notes that ineffective communication between YAEC, MYAPCo, and the NRC played an important role in the assumptions of all parties regarding the demonstration of compliance with the technical requirements of 10CFR50.46.

Demand (Section III, Item B):

Information provided to Maine Yankee in support of Cycles 14 and 15 reload analyses was not complete and accurate in all material respects regarding compliance with 10CFR50.46(a)(1).

Response:

DE&S believes that the SBLOCA analysis document transmitted to Maine Yankee (i.e., YAEC-1868) was sufficiently accurate and complete with respect to its intended audience (i.e., one knowledgeable in the LOCA analysis field such as an NRC reviewer). DE&S does note that YAEC-1868's summary is potentially misleading in that it refers to an analysis of the complete break spectrum. However, the body of the report is sufficiently detailed to accurately communicate the details of the SBLOCA analyses that were performed.

Demand (Section III, Item C):

Incorrect calculation of penetration factors, mis-application of the Alb-Chambre correlation and inadequate Quality Assurance (QA) review of report YAEC-1868 produced an SBLOCA evaluation model that overpredicted core cooling and overstated the conservatism of the evaluation model used for Maine Yankee Cycles 14 and 15.

Response:

DE&S believes that the modeling approach utilized by YAEC was reasonable and consistent with industry practices. However, DE&S does agree that a large change in penetration factors is more appropriately treated as a model revision, rather than being considered a change in model inputs as was done by YAEC. Nonetheless, DE&S believes that differing views on the modeling of physical phenomena and the treatment of inputs would have been satisfactorily resolved had an ongoing dialogue on Maine Yankee SBLOCA issues occurred with the NRC technical staff. DE&S has no reason to believe that the core cooling capability of Mane Yankee was overpredicted.

Demand (Section III, Item D):

A "Best Estimate" SBLOCA analysis prepared by YAEC was subsequently used inappropriately by Maine Yankee as part of a 10CFR50.59 analysis concerning the effects of a reduction in steam generator pressure on SBLOCA analyses.

Response:

DE&S believes that methods other than the approved Evaluation Model could have been utilized for the work undertaken, however, the limitations of use of such methods should be stated. DE&S

notes that YAEC memoranda initially mischaracterized the Maine Yankee Best Estimate model as the approved Maine Yankee Appendix K Evaluation Model. YAEC did subsequently provide MYAPCo with a draft 50.59 evaluation that was based on analyses utilizing the Appendix K Evaluation Model. DE&S also notes, that during the time frame of the referenced analyses, significant changes were underway within the NRC and industry regarding the expectations for an effective 50.59 evaluation as evidenced by publication of NRC Generic Letter 91-18 in late 1991. Nonetheless, errors occurred in mis-stating the Best Estimate model as the licensing basis analysis and in failing to state that these analyses were performed using non-NRC approved methods. Finally, DE&S expects that the effect of reduced steam generator pressure for the magnitude of steam generator tube fouling and plugging that was being evaluated would not be significant.

Demand (Section IV, Concern 1):

Regard for and adherence to NRC requirements

Response:

DE&S is 100% committed to providing products and services to NRC power reactor licensees that are unquestionably compliant with all applicable NRC requirements and the licensing basis of the nuclear station for which the work is provided. Independent assessments by DE&S of the acquired YAEC organizations show that significant improvements have been made in the treatment of NRC requirements within workplace procedures. Training has been provided to affected personnel on NRC requirements. Additionally, current work products, primarily calculations and analyses are compliant with NRC methodology guidelines and exhibit knowledge of relevant NRC requirements. A strong bias to establish safety margins through use of conservative assumptions is present. DE&S has strengthened its senior management in the Bolton office and commitment to compliance with NRC requirements by naming Mr. William H. Rasin Vice-President of Nuclear, Fuel, and Quality Assurance Services, effective March 1. Additionally, licensing functions associated with the acquired YAEC organizations will be fully integrated with existing DE&S licensing functions to strengthen the level of regulatory support provided to the acquired YAEC organizations. To provide additional assurance of quality work products that are in accordance with DE&S management expectations and in compliance with applicable NRC requirements throughout the transition and integration of the acquired YAEC organizations into DE&S, additional emphasis will be place on the DE&S independent assessment process during the transition and integration period.

DE&S firmly believes that the actions described above, in combination with actions previously implemented by YAEC, will assure that products and services provided to NRC power reactor licensees by DE&S comply with all applicable NRC requirements.

Demand (Section IV, Concern 2):

Management control and supervision over licensed activities

Response:

DE&S is 100% committed to maintaining effective management control and supervision of its safety related and NRC licensed activities. Independent assessments by DE&S of the acquired YAEC organizations show that a strong proceduralized work process that reflects NRC guidelines is in place. Significant improvements have been implemented in the deficiency reporting system and its active utilization is indicative of a workforce that openly identifies potential deficiencies. Additionally, the technical quality of work products demonstrates the effectiveness of the existing YAEC work processes in producing high technical quality. All are evidence of good management control. DE&S is verifying that its regulatory and organizational interface requirements are clearly defined and formally documented with each nuclear client formerly supported by YAEC, preventing recurrence of the organizational uncertainties that contributed to the events described in the Demand. The DE&S independent assessment function will provide DE&S management with direct feedback on the quality of work products, their conformance with DE&S management expectations, and their compliance with applicable NRC requirements throughout the transition and integration of the acquired YAEC organizations into DE&S. Other DE&S actions underway include the integration of acquired YAEC work processes and organizations into DE&S.

Corrective actions implemented by YAEC have produced a good infrastructure. Additional DE&S management actions to integrate this infrastructure into DE&S, to clearly define organizational responsibilities, emphasize strong technical oversight in the workplace, and strengthen technical leadership will provide assurance that DE&S is exerting effective management control over all safety-related and potentially safety-related activities.

Demand (Section IV, Concern 3):

Willingness of DE&S in the future to provide complete and accurate information to licensees and to the NRC.

Response:

DE&S is 100% committed to providing accurate and complete information to its power reactor licensee clients and to the NRC. The success of DE&S as a business mandates that all products and services provided to NRC power reactor licensees be accurate and complete. Independent assessments by DE&S of the acquired YAEC organizations show that the work products produced are consistently of a high technical quality and consistent with NRC guidelines and requirements. The assessments also found documentation to be adequate. In some cases, recourse to a technical document's originator was required to provide clarifications or respond to review questions. The only clearly identified trend by the DE&S assessment teams was a noticeable improvement in the quality of more recent documentation packages. DE&S has initiated a number of joint client-DE&S technical teams to review client specific documentation for adequacy to ensure that their individual business and regulatory needs are met. Additionally, DE&S is transitioning ongoing work performed by the acquired YAEC organizations to the DE&S project planning process, which requires clear definition and documentation of client requirements, methods used to ensure effective communication with the client, and training of project personnel on the requirements and provisions of the project plan.

DE&S believes that quality and accuracy of recent technical documents shows that the acquired YAEC work practices and procedures are producing quality documentation. However, older documentation may require recourse to the document's originator to provide clarifications or respond to review questions. DE&S believes that accurate and complete communication is currently being provided. Actions have been initiated with clients formerly supported by YAEC to ensure that each client's documentation needs and expectations are clearly understood and that existing documentation meets their business and regulatory needs. These actions, combined with the transition of work to the DE&S planning process and the definition of organization responsibilities described earlier, ensure that accurate and complete documentation is provided to licensees and the NRC.

Demand (Section IV, Concern 4):

Willingness and ability of DE&S to conduct activities in accordance with the Commission's requirements.

Response:

DE&S is 100% committed to conducting its work activities in a manner that complies fully with all applicable NRC requirements and the licensing basis of the nuclear station for which the work is provided. Independent assessments by DE&S of the acquired YAEC organizations show that: (1) a strong proceduralized work process that reflects NRC guidelines is in place, (2) the work products produced are consistently of a high technical quality, and (3) the work products produced are consistent with NRC guidelines and requirements. Additionally, the assessments noted that the workforce consistently demonstrates a high level of competence and knowledge within their technical areas of expertise , with an evident knowledge of industry practices and requirements. As previously described, DE&S also has underway a number of actions to further strengthen work practices and formality of operations. These actions include: (i) integration of the acquired YAEC work processes into the DE&S work processes, (ii) formally documenting organizational roles, responsibilities, and communication requirements, and (iii) strengthening management leadership. Additionally, the DE&S independent assessment function provides DE&S management with direct feedback on the compliance of work process, practices, and products with DE&S management expectations and NRC requirements.

DE&S firmly believes that using its existing work processes as a foundation, in combination with the DE&S transition actions and the corrective actions implemented by YAEC, provide assurance that DE&S work practices and conduct of work activities will comply with all applicable NRC requirements.

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1.0 INTRODUCTION

1.1 Demand for Information

On December 19, 1997, the Nuclear Regulatory Commission (NRC) issued a letter to Yankee Atomic Electric Company (YAEC) and Duke Engineering & Services, Inc.(DE&S) formally transmitting a "Demand for Information to YAEC and to Duke Engineering & Services - RE: Providing Inadequate Engineering Analyses and Materially Incomplete and Inaccurate Information to an NRC Licensee (NRC OI Report No. 1-95-050)." The Demand for Information (Demand) was issued to obtain information the NRC considered necessary to determine whether YAEC and/or DE&S should continue to provide engineering analyses, and in particular Loss of Coolant Accident (LOCA) analyses, to NRC power reactor licensees. As a result of the issues described in the Demand, the NRC requested that YAEC and/or DE&S provide the following information:

A. "An explanation why, in view of the matters set forth above [in the Demand], the NRC should permit any NRC Licensee to use the services of YAEC LOCA Group and/or DE&S, to the extent that YAEC LOCA Group was transferred to DE&S, to perform LOCA analyses or any safety-related analyses to meet NRC requirements."

B. "An explanation why the NRC should not consider the inadequate analyses, which apparently caused MYAPCo [Maine Yankee Atomic Power Company] to be in violation of NRC requirements, to be the result of willfulness, either deliberateness or careless disregard, on the part of YAEC and/or DE&S personnel."

1.2 Response Approach

DE&S review of the Demand indicates that a number of technical issues and concerns are identified. This report provides the specific information requested from DE&S, as well as a DE&S response to each of the concerns and technical issues expressed by the NRC within the Demand. DE&S has structured this report to provide a systematic and thorough response to each of these concerns and technical issues. Section 2.0 of this report describes: (1) the evaluation approach utilized by DE&S, (2) the results of the DE&S evaluation, and (3) additional follow-up actions being taken by DE&S. Section 3.0 of this report provides the DE&S response to each of the NRC concerns identified in Section IV of the Demand. These concerns are summarized as:

a. Regard for and adherence to NRC requirements,

b. Management control and supervision over licensed activities,

c. Providing complete and accurate information to licensees and to the NRC, and

d. Willingness and ability to conduct activities in accordance with NRC requirements.

Section 4.0 provides a DE&S response to each NRC technical issue identified in Section III of the Demand regarding the development and use of RELAP5YA. These technical issues are summarized as:

a. It was not possible to confirm that the limiting break had been identified and that the emergency core cooling system was capable of mitigating the most severe postulated accident for Maine Yankee Cycles 14 and 15. [Demand Section III.A]

b. Information provided to Maine Yankee in support of Cycles 14 and 15 reload analyses was not complete and accurate in all material respects regarding compliance with 10CFR50.46(a)(1). [Demand Section III.B]

c. Incorrect calculation of penetration factors, misapplication of the Alb-Chambre correlation and inadequate QA review of YAEC-1868 produced an SBLOCA evaluation model that over predicted core cooling and overstated the conservatism of the evaluation model for Maine Yankee Cycles 14 and 15. [Demand Section III.C]

d. A "Best Estimate" SBLOCA analysis, that was subsequently relied upon by Maine Yankee in connection with a 10CFR50.59 analysis concerning the effects of a reduction in steam generator pressure, was inappropriately used to determine the effects of a reduction in steam generator pressure on LOCA analyses. [Demand Section III.D]

Finally, Section 5.0 provides a DE&S response to the two NRC information requests quoted earlier. Appendix A provides a detailed description of DE&S, including a description of: (1) the company history, (2) the current scope of nuclear services DE&S provides to NRC power reactor licensees, (3) the DE&S December 1997 acquisition of YAEC assets, and (4) the DE&S quality assurance, design control, and project planning programs. Appendix B provides a summary of actions taken by YAEC in response to the Maine Yankee safety allegations, including a description of: (1) technical assessments performed, (2) the root cause assessment, (3) initial corrective actions taken, (4) longer term process improvements implemented, and (5) audits and assessments performed. Appendices C - I provide the reports of the independent assessment teams utilized by DE&S to assess the effectiveness of (1) the acquired YAEC organizations and (2) the corrective actions taken by YAEC in response to the safety allegations.

1.3 Summary Response

The DE&S assessments and management evaluation of these assessments found that the performance of, and work products and services provided by, the acquired YAEC organizations are acceptable. Consistent with previous DE&S acquisition experience, these assessments did highlight areas that require DE&S management attention to ensure a successful transition and integration of the acquired YAEC organizations into DE&S. The four most important areas highlighted are: (1) organizational roles, responsibilities and communications, (2) personnel training, (3) nuclear regulatory support, and (4) accurate and complete documentation. DE&S follow-up actions are underway in each of these areas.

The DE&S programs and procedures described within this report are revised on a routine basis. These revisions are part of routine DE&S process improvements to maintain effective programs and procedures that are in line with current industry practices and compliant with NRC requirements. Full transition and integration of the acquired YAEC organizations into DE&S will place the acquired YAEC work practices, products and services described within this report under the DE&S process improvement process. Transition of the acquired YAEC organizations into DE&S is currently scheduled for completion in July 1998.

DE&S found, based on its assessments, that the performance of, and the products and services provided by, the acquired YAEC organizations are acceptable. Meaningful improvements had been made during the past two years in the quality of YAEC procedures and the emphasis these procedures place on NRC requirements and NRC reporting. The technical quality of the work products currently being produced, as well as the professionalism and technical competence of the workforce, are consistently high. DE&S is systematically transitioning the acquired YAEC organizations to the DE&S project planning and work execution programs and procedures. The DE&S project planning process emphasizes clear definition and documentation of project-specific requirements, both technical and organizational, and training of project personnel on these requirements. DE&S has strengthened the Bolton office leadership with a proven nuclear industry executive, William H. Rasin. DE&S is also currently engaged with each of the major nuclear clients formerly supported by YAEC to ensure that roles, responsibilities, and expectations are mutually understood and documented. These actions, coupled with the nuclear commitment and experience presented by DE&S, should provide the NRC with a high degree of confidence that safety-related analyses, products and services provided by DE&S to NRC power reactor licensees comply with the principles of: (1) adherence to NRC requirements, (2) effective management control of safety-related activities, (3) accurate and complete communication with licensees and the NRC, and (4) conduct of work in accordance with NRC requirements.

A common finding or observation of the DE&S assessment teams was the high degree of professionalism and openness of the employees within the acquired YAEC organizations. All individuals were found to be honest and communicative. While in certain instances there may have been inadequate analysis, there was neither deliberateness nor careless disregard resulting from the deficiencies described in the Demand. DE&S found no willfulness on the part of the two individuals mentioned in the Demand and believes that these individuals are capable of conducting their activities in conformance with NRC requirements. DE&S also found no evidence that would question the sincerity and dedication of the acquired YAEC workforce, or would otherwise prevent DE&S activities to be conducted in full compliance with NRC requirements.

1.4 Background

New England is relatively unique in that most nuclear power stations are owned by several, relatively small utilities, many of whom have ownership in several stations. YAEC was the organizational entity formed to provide the organizational size and qualifications necessary to design and operate these nuclear stations. In fact, YAEC was the original NRC Licensee for the Maine Yankee, Vermont Yankee and Yankee Rowe nuclear stations. As a result of this multi-plant, multi-ownership situation, the organizational relationships between YAEC and the

utilities was derived more from common nuclear history and experience rather than explicitly defined requirements. In particular, with respect to Maine Yankee, MYAPCo became the NRC Licensee in 1981. With this change, YAEC was no longer the primary interface with the NRC. Although YAEC continued as the primary provider of engineering services to Maine Yankee, many organizational relationships and responsibilities were not explicitly defined at this point. It was within this organizational framework that YAEC undertook the development, NRC approval, and application of RELAP5YA throughout the 1980's and 90's.

On December 1, 1995, the Union Of Concerned Scientists (UCS) transmitted a letter to the State Nuclear Safety Advisor of the Maine State Planning Office stating that they had received anonymous documentation, "....purportedly from a longtime employee of the Yankee Atomic Electric Company...." alleging that the management at Maine Yankee deliberately falsified reports to the NRC in order to receive approval of an increase in the reactor's maximum allowable power level. The UCS letter further stated that the anonymous documentation alleged that management officials [at Yankee Atomic] manipulated computer calculations to avoid disclosing that the emergency core cooling systems at the Maine Yankee plant were inadequate to prevent overheating of the reactor fuel following a SBLOCA. The letter requested that Maine Yankee not be permitted to resume operation until a factual investigation of the anonymous allegations was completed. The NRC received these allegations on December 4, 1995.

The NRC promptly dispatched an Assessment Team to YAEC headquarters in Bolton, Massachusetts on December 11-14, 1995 to conduct a technical review of the allegations. The NRC Assessment Team was accompanied by representatives of the State of Maine. On December 18, 1995 the NRC held a public meeting in Rockville, Maryland with MYAPCo and YAEC to discuss the findings of the technical review and to seek additional information. As a result of the technical review and subsequent evaluations, the NRC issued on January 3, 1996 a "Confirmatory Order Suspending Authority For and Limiting Power Operation and Containment Pressure, and Demand for Information" to MYAPCo. The order was based, in part, on the NRC's determination that Maine Yankee had not applied a computer code that was proposed to demonstrate compliance with the Emergency Core Cooling System (ECCS) requirements of 10CFR50.46 in a manner that conformed to the requirements of 10CFR50, Appendix K, nor to the conditions specified in the staff's safety evaluation report dated January 30, 1989. Specifically, the confirmatory order and demand for information stated that MYAPCo allegedly could not demonstrate that the RELAP5YA code would reliably calculate the peak cladding temperature for all break sizes in the small-break LOCA spectrum for Maine Yankee, nor had MYAPCo submitted the justification for the code options selected and other justifications and sensitivity studies to satisfy conditions in the NRC staff's safety evaluation.

On January 10, 1996, MYAPCo provided an initial response to the Demand for Information. On January 23, 1996, MYAPCo responded to the Confirmatory Order. This letter also indicated that MYAPCo and YAEC had performed extensive technical evaluations of the issues raised by the UCS allegations. MYAPCo provided reports to the NRC on January 22, 1996 of two independent teams formed by MYAPCo and YAEC. On February 2, 1996, MYAPCo provided a schedule for responding to the remaining items in the NRC's January 3, 1996 Demand for Information. Included in this schedule was completion of

an independent SBLOCA analysis to be performed by Siemens. The results of this analysis were submitted to the NRC on April 23,1996 by MYAPCo.

In response to the initial NRC investigations, an inquiry conducted by the NRC Office of Inspector General (OIG), and concerns expressed by the Governor of Maine, the NRC Chairman initiated a separate special investigation during the summer of 1996. This investigation was performed by an independent team of NRC experts and consultants with the objective of verifying YAEC engineering activities other than those related to SBLOCA analysis. The scope of this Independent Safety Assessment Team (ISAT) as stated in their report, dated October 7, 1996 was:

> "On May 31, 1996, the staff was directed to perform an independent evaluation of Maine Yankee's safety performance. The overall goals of the independent safety assessment were to: (1) independently assess the conformance of Maine Yankee Atomic Power Station (MYAPS) to its design and licensing bases including appropriate reviews at the site and corporate offices; (2) independently assess operational safety performance giving risk perspectives where appropriate; (3) evaluate the effectiveness of licensee self-assessments, corrective actions, and improvement plans; (4) determine the root cause(s) of safety-significant findings and draw conclusions on overall performance."

In general, the ISAT concluded: "The quality of engineering work was mixed but considered good overall. Strengths were noted in the capability and experience of the engineering staff, day-to-day engineering support of maintenance and operations, in the quality of most calculations, and in the routine use and application of analytic codes" On December 10, 1996, MYAPCo responded to the ISAT report with a corrective action plan to address all of the issues identified in the ISAT report.

Continued review and investigation led the NRC to conclude by December, 1997 that by YAEC's preparation and approval of the RELAP5YA SBLOCA analysis and the WREM Large Break LOCA (LBLOCA) analysis, and by YAEC's preparation and approval of the Core Performance Analysis Reports (CPARs) used to support Cycle 14 and Cycle 15 operation of Maine Yankee, YAEC caused MYAPCo to be in apparent violation of 10CFR50.46(a)(1). Specifically, the NRC stated that the RELAP5YA SBLOCA analysis and the WREM LBLOCA analysis, singly or combined, were not capable of acceptably calculating emergency core cooling system performance for the portion of the break spectrum between 0.35 ft^2 and at least 0.6 ft^2. Furthermore, the NRC stated that it was not possible to confirm that the limiting break had been identified and that the emergency core cooling system was capable of mitigating the most severe postulated accident. Moreover, the NRC concluded that YAEC provided MYAPCo with information that was not complete and accurate in all material respects regarding this noncompliance with 10CFR50.46(a)(1), and thus caused MYAPCo to apparently violate 10CFR50.9(a) by maintaining CPARs which contained information which was not complete and accurate in all material respects in connection with MYAPCo's Cycle 14 and Cycle 15 reload analyses.

Additionally, the NRC concluded that as a result of incorrect calculation of penetration factors, misapplication of the Alb-Chambre correlation, and inadequate QA review of YAEC-1868, YAEC caused MYAPCo to rely on an unacceptable SBLOCA evaluation model which over predicted core cooling and overstated the conservatism of the evaluation model for Cycle 14 and Cycle 15 in apparent violation of 10CFR50.46(a)(1). Finally, the NRC concluded that by its use of an unacceptable "Best Estimate" SBLOCA analysis to determine the effects of a reduction in steam generator pressure on LOCA analyses, YAEC caused MYAPCo to apparently violate 10CFR50.46(a)(1). Specifically, MYAPCo relied upon this unacceptable "Best Estimate" SBLOCA evaluation model to calculate ECCS cooling performance in connection with a 10CFR50.59 analysis concerning the effects of a reduction in steam generator pressure.

As a result of these conclusions, the NRC issued the Demand on December 19, 1997 to both DE&S and YAEC. In addition, concurrent with the issuance of the Demand, MYAPCo was notified by the NRC of apparent violations associated with performing inadequate LOCA analyses and providing inaccurate information to the NRC.

1.5 DE&S Perspective

DE&S, a wholly owned affiliate of Duke Energy Corporation (Duke), is committed to the highest standards of integrity and technical quality upon which its clients and the Nuclear Regulatory Commission (NRC) can rely without reservation. DE&S fully understands and serves its NRC Licensee clients in accordance with these basic principles:

a. adherence to NRC and industry requirements and standards,

b. effective management control of safety-related activities,

c. accurate and complete communication with clients and the NRC, and

d. conduct of work in accordance with NRC and industry requirements.

Successful compliance with these principles has contributed, in large part, to the success of Duke's nuclear program. Just as importantly to DE&S, compliance with these principles in providing technical products and services to NRC Licensees is fundamental to the success of DE&S as a business. A detailed description of DE&S is provided in Appendix A, including: (i) the company history, (ii) the current scope of nuclear services DE&S provides to NRC power reactor licensees, (iii) the DE&S December 1997 acquisition of YAEC assets, and (iv) the DE&S quality assurance, design control, and project planning programs.

Prior to the December 1997 acquisition of a portion of its assets by DE&S, YAEC had conducted a number of assessments and implemented a number of corrective actions designed to address the concerns and issues raised by the NRC during, and subsequent to, its investigations of the December 1995 safety allegations regarding Maine Yankee. DE&S was aware of these allegations and the NRC's resulting investigations. DE&S conducted the necessary organizational assessments of YAEC during the "due diligence" phase of the acquisition. Additionally, DE&S had planned to conduct a number of more detailed

independent assessments of work processes, activities, and products after the acquisition as an element of the transition and integration of the acquired YAEC organizations into DE&S. These DE&S assessments were accelerated and expanded to ensure that DE&S was fully responsive to the NRC's December 1997 Demand.

The focus of post-acquisition actions by DE&S is straight forward - ensure that the new DE&S organizations are providing quality technical products and services that both comply with the principles listed above and meet the needs of DE&S clients and the NRC. This demonstration is essential for two reasons: (1) to ensure successful continuity of ongoing work with quality products and services and (2) to allow a systematic transition from the acquired YAEC work processes and procedures to DE&S work process and procedures. Regarding continuity of quality products and services, the new DE&S organizations are currently working in accordance with pre-existing YAEC work processes, programs, and procedures. This provides the most efficient means of assuring continuity of ongoing work at the time of the acquisition and continued compliance with governing regulatory commitments and requirements. Showing that the acquired processes and procedures reliably produce quality products and services ensures that needs of clients and the NRC are currently being satisfied. Regarding transition of the acquired work processes and procedures, these will be systematically assimilated into existing DE&S programs and procedures. By design during the transition period, decreasing quantities of on-going work will continue to be performed under the acquired YAEC procedures. This approach permits assimilation of these processes into DE&S programs and procedures in a systematic manner and reduces the potential disruption of ongoing work associated with an accelerated work process transition. Transition of the acquired YAEC organizations into DE&S is currently scheduled for completion in July 1998.

2.0 DE&S EVALUATION AND ACTIONS

2.1 Evaluation Approach

DE&S review of the Demand indicates that the NRC has a number of concerns. In order to ensure a fully responsive reply to the Demand, DE&S concluded that a number of objectives had to be demonstrated. These objectives are to assure that:

a. The underlying cause(s) of events that led to the Demand have been identified and appropriate corrective actions have been successfully implemented,

b. Work products and practices exhibit an understanding of and adherence to governing NRC and industry requirements,

c. Work products and practices exhibit management control over safety-related activities,

d. Work product documentation is complete and accurate, and

e. Work products are technically correct and defensible.

Additionally, DE&S has to make a determination regarding the "willfulness" of actions and events summarized in the Demand and address the technical issues stated in the Demand regarding safety analyses performed for the Maine Yankee plant. These specific technical issues are:

a. It was not possible to confirm that the limiting break had been identified and that the emergency core cooling system was capable of mitigating the most severe postulated accident for Maine Yankee Cycles 14 and 15. [Demand Section III.A]

b. Information provided to Maine Yankee in support of Cycles 14 and 15 reload analyses was not complete and accurate in all material respects regarding compliance with 10CFR50.46(a)(1). [Demand Section III.B]

c. Incorrect calculation of penetration factors, misapplication of the Alb-Chamber correlation and inadequate QA review of YAEC-1868 produced a SBLOCA evaluation model that over predicted core cooling and overstated the conservatism of the evaluation model for Maine Yankee Cycles 14 and 15. [Demand Section III.C]

d. A "Best Estimate" SBLOCA analysis, that was subsequently relied upon by Maine Yankee in connection with a 10CFR50.59 analysis concerning the effects of a reduction in steam generator pressure, was inappropriately used to determine the effects of a reduction in steam generator pressure on LOCA analyses. [Demand Section III.D]

All of these objectives were achieved by systematically subjecting the acquired YAEC organizations to a series of horizontal and vertical independent assessments. This process of systematic horizontal and vertical organizational assessments by DE&S was previously planned as an integral element of the transition and integration of the acquired YAEC organizations

into DE&S. The issuance of the Demand prompted DE&S to both accelerate and expand the scope of these assessments in order to be fully responsive to the Demand.

The independent assessments included an examination and evaluation of both the technical and process aspects of work performance. A significant focus was placed on LOCA and associated safety analyses performed by the Nuclear Engineering Department (NED), with detailed technical assessments conducted of these work products.[Note: NED contains what was formerly the YAEC LOCA Group.] Additionally, assessments were conducted of: (1) engineering and technical work practices and products other than those related to LOCA analyses, (2) the Quality Assurance assessment program, and (3) corrective actions initiated by YAEC in response to results of their root cause assessments. The specific DE&S assessment activities performed were:

a. Maine Yankee SBLOCA Analysis Assessment - This effort was an independent technical review of the technical issues summarized in the Demand.[Appendix C]

b. Personnel Behavior Assessment - This effort was an independent legal review of the "willfulness" of the actions and events summarized in the Demand. [Appendix D]

c. Root Cause and Corrective Action Assessment - This effort was an independent review of the YAEC 1996 Root Cause Assessment and YAEC's 1996-1997 Corrective Actions. [Appendix E]

d. Engineering Process Assessment - This effort was an independent review of engineering/technical work practices used and work products produced during the past 3 years, other than those associated with LOCA analyses. [Appendix F]

e. Quality Assurance Assessment - This effort was an independent review of YAEC QA audits performed during the past 3 years. [Appendix G]

f. Vermont Yankee Safety Analysis Assessment - This effort was an independent technical review of: (i) Basis for Maintaining Operation (BMO) documents, (ii) Cycle 20 reload analyses, and (iii) containment analyses. [Appendix H]

g. Seabrook Safety Analysis Assessment - This effort was an independent technical review of radiological consequence and RELAP5YA non-LOCA transient analyses. [Appendix I]

These assessments were conducted by individuals who are knowledgeable in, and independent of, the functional areas reviewed. Assessment team members were obtained from DE&S, Duke, subcontractors, and independent consultants. A listing of these individuals and their company affiliations is provided in Table 2-1. The assessment activities included (1) interviews with knowledgeable technical and management personnel, (2) reviews of governing work activity procedures, and (3) technical reviews of work products, typically calculations. Throughout the course of these assessments, DE&S management met daily with the assessment teams to review progress and potential findings. When a specific weakness or potential deficiency was identified, it was entered into the acquired YAEC Condition Report

System. Additionally, affected stations (e.g., Seabrook and Vermont Yankee) were notified of the weakness or potential deficiency for their own internal evaluation and follow-up.

The findings and conclusions of each assessment are summarized in the Appendices, as indicated previously in Section 2.1. DE&S management's evaluation of the overall findings and conclusions of the assessments is provided in below Section 2.2.

2.2 Evaluation Results

Results of the independent assessment teams were integrated with DE&S management observations during and subsequent to the acquisition's "due diligence" assessments to produce an evaluation of the acquired YAEC organizations with respect to each evaluation objective listed in Section 2.1. This DE&S management evaluation found that the current performance of, and work products and services provided by, the acquired YAEC organizations are acceptable. Consistent with previous DE&S acquisition experience, the DE&S assessments did identify areas requiring management attention to ensure a successful organizational transition and integration. A discussion of these areas and planned DE&S follow-up actions is provided in Section 2.3.

2.2.1 Root Cause and Corrective Actions Validation

Evaluation Objective:

Assure that the underlying cause(s) of events that led to the Demand have been identified and appropriate corrective actions have been successfully implemented.

DE&S Evaluation:

Three (3) underlying causes were identified by YAEC as a result of an assessment performed in early 1996: (1) definition of organizational roles and responsibilities was less than adequate, (2) training of YAEC personnel was less than adequate, and (3) clarity and specificity of procedures was less than adequate. Each cause and its associated corrective actions is addressed separately below.

2.2.1.1 Organizational Responsibilities, Roles, and Communications

The 1996 YAEC root cause assessment determined that the division of responsibilities/ownership and roles of organizations performing, controlling, administering, and managing activities for the Maine Yankee plant were not always completely and clearly defined or understood by all parties involved in or impacted by the activity. DE&S concurs with this assessment. As described in Appendix E, many YAEC employees also believed that the relationship between YAEC and MYAPCo was subject to varying definitions, depending on what relationship was in the best commercial interest of Maine Yankee for a given situation. This less than adequate definition of organizational roles and responsibilities clouded, and thus hindered, effective communications between YAEC and MYAPCo and with the NRC.

The DE&S assessments found that personnel currently have a good understanding of their individual and work group roles and responsibilities at the working level. This working level understanding of roles and responsibilities was validated through the technical assessment teams and the process assessment teams. In particular, as described in Appendix E, the Root Cause and Corrective Action Assessment determined that corrective actions were initiated by YAEC to address definition of roles and responsibilities between YAEC and MYAPCo. It should be noted that the subsequent 1997 decision by MYAPCo to permanently shutdown Maine Yankee rendered many of these initiatives moot. Nonetheless, these corrective actions were focused primarily on MYAPCo and did not fully address roles and responsibilities with other plants supported by YAEC. Additionally, the Engineering Process Assessment, which by design took a broader view of the organization than the relatively narrow focused technical assessment teams, determined that the employees within the acquired YAEC organizations have a limited overall understanding of the current DE&S organization. Furthermore, on-going interactions with DE&S clients formerly supported by YAEC indicate that additional clarifications regarding expectations and interface requirements with DE&S are warranted. (It should be noted that client representatives participated in an oversight role in these reviews, but were not interviewed as an element of the DE&S independent assessments.)

It is recognized by both employees within the acquired YAEC organizations and DE&S clients who were formerly supported by YAEC, that the acquisition of YAEC assets by DE&S has created an new organizational relationship. This new relationship is one of a client and a subcontractor, which removes the uncertainties caused by the co-ownership relationship that existed during the events described within the Demand. In fact, DE&S and its clients formerly supported by YAEC are in the process of negotiating contract modifications that more clearly define these respective roles, responsibilities, and expectations. As these contract modifications are enacted, DE&S workplace procedures and programs will be revised to conform to the specified contract requirements.

2.2.1.2 Company Training

The 1996 YAEC root cause assessment determined that personnel at YAEC lacked formal company training and retraining in areas that are essential to performance of their day-to-day tasks. Informal company training and retraining referred, in part, to the read and sign training approach for day-to-day work procedures. Although read-and-sign is a useful training tool, it is not alone sufficient to ensure effective personnel training. Formal classroom instruction is at times necessary to ensure a complete understanding of codes, standards, procedures, regulations and management expectations. DE&S concurs with this assessment.

The DE&S technical and process assessment teams found that the employees within the acquired YAEC organizations demonstrate a high level of competence and knowledge within their respective technical areas of expertise. The technical expertise of these employees was acknowledged consistently as a strength and will be discussed later. The Engineering Process Assessment and the Quality Assurance Assessments, in particular, reviewed training records for individuals originating, reviewing and approving documents that were reviewed as part of the assessment activities. These assessments found that personnel were appropriately qualified to perform their assigned functions. However, the Root Cause and Corrective Action Assessment determined that actions taken by YAEC to improve technical training were not complete and systematic. Although additional training had been introduced into the YAEC organization (specifically regarding NRC regulations and reporting requirements) and procedures governing training requirements were in place, evidence of a systematic commitment to improve the quality of personnel training was not found. While it was determined that more emphasis on personnel training is warranted, the cumulative findings of the assessment teams indicate that the employees within the acquired YAEC organizations are qualified to perform their work assignments.

2.2.1.3 Procedure Quality

The 1996 YAEC root cause assessment determined that procedures used for controlling the development of analyses were weak in defining important processes that appeared to be driven by personnel knowledge rather than by procedural guidance. These procedures did not require identification of the effects of analyses on licensing commitments or design basis documents. YAEC recognized that governing analysis procedures needed to be sufficiently clear with respect to ensuring consideration of regulatory requirements, effective controls on the use of unverified data, compliance with QA requirements, updating of affected design basis and/or licensing documents. DE&S concurs with this assessment.

The DE&S technical and process assessment teams found that current workplace procedures are effective, are being properly implemented, and reflect current industry standards and practices. Several assessments, in particular the Engineering Process and the Root Cause and Corrective Action Assessments, noted the presence of management initiatives to continuously improve technical procedures. The Root Cause and Corrective Action Assessment did express concern regarding the relatively narrow focus of corrective actions on MYAPCo, instead of more globally addressing all YAEC clients. Nonetheless, the Assessment concluded that the root cause had been adequately addressed, and that the specific concerns of self-assessment, analysis control, and NRC reporting have been addressed very well. These corrective actions and improvements were programmatic and applicable to all YAEC clients.

The Engineering Process Assessment noted that the transition from YAEC to DE&S is creating some concern among former YAEC employees regarding the impact full integration into DE&S will have on their procedures. This concern is normal as staff face a transition from procedures with which they are familiar to a set of procedures that may be very different in form and content. In fact, DE&S is aware of this and has implemented a systematic transition from YAEC to DE&S work practices, utilizing a process that ensures assimilating the best elements of YAEC procedures and work practices into the DE&S procedures and work practices. DE&S recognizes the importance of the transition from one procedure set to another, as well as the benefit of retaining the valuable lessons learned that have been incorporated into the YAEC procedures.

2.2.2 *Adherence to NRC Requirements*

Evaluation Objective:

Assure that current work products and practices exhibit an understanding of and adherence to governing NRC and industry requirements.

DE&S Evaluation:

Numerous work products and procedures were reviewed and interviews conducted by the assessment teams to address the stated evaluation objective with respect to the adequacy of the work products, governing procedures and personnel conducting the work. A common finding or observation from both the technical and process assessment teams was that the workforce consistently demonstrates a high level of competence and knowledge within their technical areas of expertise. Questions are answered accurately and correctly without hesitation. Typically, detailed explanations are provided without recourse to reference material. Available references and backup material are promptly provided upon request. Knowledge of industry practices and safety consciousness is evident.

Actions taken to improve the quality of technical procedures, especially with respect to treatment of NRC requirements are described in Subsection 2.2.1.3. Another general finding or observation of both the technical and process assessment teams is that work products and their governing procedures are consistent with NRC requirements and generally accepted industry practices. In particular, a broad sampling of safety analyses found that analysis methods used follow guidance provided by the NRC's Regulatory Guides and/or Standard Review Plan. Technical analyses and documents are consistent with similar analyses and documents throughout the industry.

Assessment results show that the acquired YAEC organizations contain quality people producing quality products in accordance with NRC and industry guidelines. However, there are indicators that continued management attention is warranted. The circumstances surrounding the development, licensing, and utilization of RELAP5YA for the Maine Yankee SBLOCA analysis did not clearly display complete understanding

of governing NRC regulations. (The DE&S evaluation of the RELAP5YA issues are discussed in Section 4.0 and Appendix C of this report.) Isolated situations were also identified by the technical assessment teams where a document's originator was either uncertain with respect to the licensing basis within which the product was to be used or not sensitive to the necessary timeliness and documentation requirements to demonstrate compliance with a plant's licensing basis or an NRC requirement. Another indication mentioned by the Root Cause and Corrective Action Assessment refers to the limited nature of organized licensing support provided to the technical workforce. The timely involvement of licensing professionals to either determine or respond to questions regarding the current licensing basis requirements governing a particular analysis or project is of significant value to technical personnel.

As described above, procedures have been improved, training has been provided regarding NRC regulations, and work products do display an awareness of and adherence to NRC technical requirements and industry standards. Therefore, the fundamental and necessary components are in place. Continued management attention will ensure maintenance of a work environment that is fully sensitized to compliance with both the letter and spirit of NRC requirements.

2.2.3 *Management Control*

Evaluation Objective:

Assure that current work products and practices exhibit management control over safety-related activities.

DE&S Evaluation:

Evidence of management control over safety-related activities is generally sought by examination of several avenues: (1) clearly defined and communicated management expectations regarding organizational policies and work practices, (2) workforce understanding and compliance with management expectations and policies, (3) the formality of operations exhibited by the organization, (4) willingness of the workforce to identify problems to management, and (5) management visibility through out the workplace in a manner that reinforces management's priorities and expectations.

The Engineering Process, the Quality Assurance, and the Root Cause and Corrective Action Assessments each reviewed workplace procedures for conformance with applicable NRC requirements and industry standards. The collective findings of these assessments were that existing procedures properly incorporated applicable NRC and industry requirements, and were consistent with industry practices. Thus, one indication that management control is evident is the fact that management expectations regarding the manner in which work is executed is appropriately and effectively contained within workplace procedures.

Workforce understanding of management expectations is generally demonstrated through the quality of the work products. As mentioned earlier and discussed later in

Subsection 2.2.5, both the process and technical assessment teams generally found that work products were of a high technical quality and were performed in accordance with appropriate workplace procedures. The quality of technical work products provides evidence of employee understanding of and compliance with management expectations as they are expressed within workplace procedures.

As discussed previously, the early relationships that existed between YAEC and the NRC Licensees supported by YAEC were not always strictly defined or documented in detail. This approach supported a strong sense of comraderie and esprit de corp that in some ways served YAEC and its clients well, but also spawned the uncertainties regarding organizational roles, responsibilities, and communications that played such a significant role in the events described within the Demand. Corrective actions taken by YAEC inserted more discipline into the YAEC-MYAPCo relationship. However, the organization is going through another transition with the DE&S acquisition. Nonetheless, quality work products and services are being provided. DE&S management recognizes that the changed organizational relationship between the acquired YAEC organizations and the nuclear clients they support, combined with the DE&S commitment to full compliance with NRC requirements, warrants an increased management focus on discipline of operations.

Appendix B, Section B.4, describes the recent development of and improvements in the YAEC reporting systems for deficiencies and employee concerns. The growth in usage of the deficiency reporting system is illustrated. A "healthy" number of deficiency reports is a positive sign which can be linked to the questioning attitude of the workforce and their ability and desire to identify potential problems without fear of retribution. Additionally, another common observation of both the process and technical assessment teams was the openness and honesty of the workforce throughout the assessment interviews. An important aspect of problem identification pointed out within the Root Cause and Corrective Action Assessment is the concern about the degree of a questioning attitude present with respect to work assignments. A specific example was found where a calculation was requested by a client and performed. However, the calculation was not sufficient to fulfill the needs of the client. Subsequent to completion and transmittal of the calculation, the discrepancy between the scope of the calculation and its intended use was internally identified and a deficiency report promptly initiated. It is a positive point that a questioning attitude within the organization discovered and reported the deficiency, albeit after completion and transmittal of the calculation.

Finally, management within the acquired YAEC organizations is visible throughout the work place, as evidenced by active participation in DE&S assessment teams' entrance and exit meetings (as well as the daily team status briefings), appropriate participation in technical meetings associated with ongoing client work, and appropriate participation in client interface meetings. Additionally, the Quality Assurance Assessment observed that active management participation in QA surveillance and audit oversight activities was evident throughout the past three years. Thus, management is visible in the workplace, particularly with respect to activities affecting quality.

2.2.4 Accurate and Complete Documentation

Evaluation Objective:

Assure that current work product documentation is complete and accurate.

DE&S Evaluation:

A broad based approach was taken to assess the accuracy and completeness of documentation. Work product documentation reviewed included calculations, analyses, procedures, training records, technical documents, correspondence, and software documentation. The guideline used to judge the adequacy of analyses, calculations and other technical reports and documents was whether the documents were sufficiently detailed as to purpose, method, assumptions, inputs, references, and units that a technically qualified individual could understand the documents and verify their adequacy without recourse to the originator.

Overall, the findings and observations of the assessment teams were mixed regarding documentation quality. Some calculations were considered by reviewers to be excellent documents. In other cases, documents were difficult to follow, although in every situation where this occurred, the originator was able to satisfactorily respond to the reviewer's questions. Nevertheless, the above-stated document quality guideline was not met in these cases. Additionally, situations were identified where training and/or qualification records for personnel participating either in the production of technical documents or surveillance of technical activities were either missing or incorrect. These situations were not widespread and were typically quickly remedied. One example is the Quality Assurance Assessment finding regarding missing or incorrect documentation supporting the training and qualifications of auditors and technical specialists in QA audits and surveillances. However, the assessment did conclude that the acquired YAEC QA program assessment process was of high quality and otherwise compliant with regulatory requirements. Another documentation concern identified by one of the technical assessment teams was the timeliness of completing formal documentation packages, particularly with respect to documentation of analyses performed to support operability determinations. Given the scope of the DE&S assessment teams, it is difficult to identify any clear trends based on the derived data, other than the assessments did observe a noticeable improvement in quality from older to more recent documentation packages.

In summary, documentation was found to be accurate, but in some cases discussions with a technical document's originator were necessary to provide clarifications or respond to review questions. Therefore, utilizing the guideline defined above, more focus is required to enhance work product documentation quality. DE&S is initiating actions with each client to ensure that their documentation needs and expectations are being met.

2.2.5 *Technical Quality*

Evaluation Objective:

Assure that current work products are technically correct and defensible.

DE&S Evaluation:

Numerous analyses, calculations, and technical documents were reviewed by the assessment teams to judge the technical adequacy of these work products. A common finding or observation from both the process and technical assessment teams was that no significant technical errors were identified in completed analyses. Minor problems were periodically identified, however, particularly in analyses that were "in process" and had not been reviewed and completed. It was also generally noted that an emphasis on the application of conservative approaches and/or assumptions is clearly evident. Examples were identified of analyses that were described as "state of the art" or "one of the best seen." Thus, it is clear from the assessment results that the technical work products produced by the acquired YAEC organizations are both technically correct and defensible.

2.3 Follow-up Actions

Identification of follow-up actions by DE&S was accomplished through a DE&S senior management review of the evaluation results in the context of DE&S management expectations and standards, as well as those of DE&S clients, the commercial nuclear industry, and the NRC. These follow-up actions are discussed in the Subsections below.

As previously stated, prior to receipt of the Demand, DE&S had planned to conduct assessments of the acquired YAEC organizations in the normal course of transitioning and integrating the acquired YAEC organizations into DE&S. These assessment results would not only assist DE&S management in constructing an effective transition and integration plan, but also would highlight functional and/or organizational items that warrant particular DE&S management attention. Previous DE&S acquisitions have shown that these "highlighted" results from transition and integration assessments typically have been items that are either: (1) vulnerabilities requiring attention to ensure a successful transition or (2) areas where implementation of DE&S management expectations and work practices require attention to ensure successful integration. Although the Demand prompted DE&S to accelerate and expand the scope of its planned assessments, the fundamental purpose and role of these assessments remained unchanged, namely a set of DE&S management tools to ensure successful transition and integration of the acquired YAEC organizations into DE&S.

DE&S management evaluation of assessment results (reference Section 2.2) found that the performance of, and work products and services provided by, the acquired YAEC organizations are acceptable (e.g., meeting NRC requirements). Consistent with previous DE&S acquisition experience, the DE&S assessments did identify areas requiring management attention to ensure a successful organizational transition and integration. The follow-up actions described below address the four most important areas identified as requiring management

attention to ensure successful transition and integration of the acquired YAEC organizations into DE&S. These four areas are: (1) organizational roles, responsibilities and communications, (2) personnel training, (3) nuclear regulation support, and (4) accurate and complete documentation.

Transition of the acquired YAEC organizations into DE&S is currently scheduled for completion in July 1998. It should also be noted that existing DE&S programs and procedures already address many areas identified by YAEC in 1996 as requiring improvement. One example of this is the Employee Concerns Program. However, one DE&S transition practice is to incorporate the experience and lessons learned of acquired organizations into DE&S programs and procedures. This practice allows DE&S to benefit from YAEC's experiences and ensures that the process improvements previously implemented by YAEC are either addressed by existing DE&S procedures or are incorporated into DE&S procedures.

The DE&S programs and procedures described within this report are revised on a routine basis. These revisions are part of routine DE&S process improvements to maintain effective programs and procedures that are in line with current industry practices and compliance with NRC requirements. Full transition and integration of the acquired YAEC organizations into DE&S will place the acquired YAEC work practices, products and services described within this report under the DE&S process improvement process.

2.3.1 *Organizational Responsibilities, Roles and Communications*

As part of the corrective actions in response to the YAEC root cause assessment, actions were taken by YAEC and MYAPCo to better define roles, responsibilities and communication requirements between YAEC and MYAPCo. However, a global effort to ensure clear definitions and understanding with all YAEC clients was not specifically undertaken. The DE&S acquisition of YAEC organizations introduces both the need and the opportunity to define organizational roles, responsibilities and communication requirements between DE&S and nuclear stations formerly supported by YAEC. To ensure a successful transition, DE&S actions are warranted to establish roles, responsibilities, relationships, and communication requirements that are properly defined and documented with each of the nuclear clients formerly supported by YAEC. This action will also prevent recurrence of one of the identified YAEC root causes. Discussions are underway with each of these nuclear clients to ensure a clear understanding of roles, responsibilities and communication requirements. The results of these discussions will be formally documented, with DE&S procedures revised as appropriate to reflect these understandings. Additionally, these understandings will be formally communicated to employees and training conducted on any resulting procedure revisions.

A Management Directive has been issued to the acquired YAEC organizations formally communicating DE&S management expectations regarding interactions with DE&S clients. This Directive addresses three items: (1) feedback from the DE&S assessments regarding organizational roles and responsibilities, (2) DE&S management's view of the new organizational relationships created by the DE&S acquisition, and (3) DE&S management expectations regarding the role of DE&S as a service provider in

communicating with and supporting commercial nuclear clients. A particular point to be noted as an element of DE&S management expectations on providing support to nuclear clients is the necessity of understanding the ultimate use of a requested analysis, calculation, or design. Routinely obtaining and documenting this understanding will ensure that the client's "real" problem or need is being addressed and that appropriate regulatory requirements are applied within the work product in a manner commensurate with its intended use.

In addition to addressing organizational relationships, an equally important relationship where ensuring clear definition is required at the individual project level. The DE&S project planning process is summarized in Appendix A, Section A.3.3. This process requires, in part, the identification of responsibilities and communication requirements for each project. The DE&S project planning process also requires that project personnel be trained on the requirements contained within the project plan. This training requirement includes project-specific roles, responsibilities, and communication requirements. Therefore, it is important to DE&S and its clients formerly supported by YAEC to promptly and systematically transition all on-going work performed by the acquired YAEC organizations to the DE&S project planning system. Transition of the acquired YAEC organizations into DE&S is currently scheduled for completion in July 1998.

The complete transition of ongoing work and new work performed by the acquired YAEC organizations to the DE&S project planning process, in combination with the corporate level actions described above, will ensure that responsibilities, roles, and communication requirements are clearly defined and documented at both corporate and individual project levels.

2.3.2 *Personnel Training*

Corrective actions were taken by YAEC in response to the 1996 root cause assessment to address immediate training deficiencies, particularly with respect to NRC reporting requirements. However, the need for a systematic training function and incomplete YAEC corrective actions are items that warrant DE&S management focus during transition and integration of the acquired YAEC organizations into DE&S. As a result, systematic integration of the acquired YAEC training functions into DE&S training processes is an important transition objective. Full integration will permit timely performance of job task analyses of the nuclear related work performed by the acquired YAEC organizations, allowing DE&S to systematically define each organization's training requirements. Overlaying the results of this analysis on the existing training program will permit the systematic identify training deficiencies and permit development personnel training matrices, from which individual training schedules will be developed.

2.3.3 *Nuclear Regulatory Support*

DE&S management must set and enforce the expectations and standards regarding regulatory compliance. The DE&S evaluation found that existing procedures have been

revised to appropriately address applicable NRC requirements and that YAEC had provided personnel trainin on NRC reporting requirements. A strengthened training program will further ensure workforce understanding of NRC requirements. Nonetheless, a continual message from DE&S management to the workforce reinforcing the importance of regulatory compliance is appropriate. DE&S has selected William H. Rasin to become DE&S Vice-President of Nuclear, Fuel, and Quality Assurance Services effective March 1, 1998. Mr. Rasin's organization consists of these former YAEC departments: Nuclear Engineering, Fuel Management, and Quality Assurance. Mr. Rasin has an extensive background with Duke, and more recently with the Nuclear Energy Institute (NEI), in managing complex nuclear safety and regulatory issues. Appendix J provides a copy of Mr. Rasin's resume. Mr. Rasin's experience and presence in the DE&S Bolton office, will provide solid DE&S management emphasis to the Bolton office workforce in this area.

Another item requiring management focus during transition and integration is to ensure that appropriate nuclear licensing support is effectively integrated into ongoing work performed by the acquired organizations. The actions described in Subsection 2.3.1 provide means of identifying NRC regulatory support requirements at both the project and corporate level. Appendix A, Section A.1.5 provides a brief summary of DE&S licensing expertise.

2.3.4 *Accurate and Complete Documentation*

The variability in the quality of technical documents warrants increased DE&S management attention. The quality of recent technical documents indicates that work practices and procedures are producing work products and services supported by quality documentation. To further ensure the adequacy of supporting technical documentation, DE&S has initiated a number of joint client-DE&S technical teams to review client specific documentation needs and expectations. The purpose of these teams is to determine whether existing technical documentation is sufficiently self-contained to adequately meet their business and regulatory needs. DE&S procedures will be revised as appropriate to reflect these documentation needs and expectations. Employees will be trained on resulting procedure revisions.

A Management Directive has also been issued to the acquired YAEC organizations formally communicating two items: (1) the findings of the DE&S independent assessments regarding documentation quality and (2) DE&S management expectations regarding completeness of technical documentation. Additionally, the DE&S independent assessment process provides direct feedback to DE&S senior management regarding the effectiveness of the organization's implementation of management expectations and the maintenance of quality and compliant work products. Particular management emphasis will be placed on independent assessments throughout the transition from YAEC programs and procedures to DE&S programs and procedures to ensure that the acquired YAEC organizations continue to provide quality products and services throughout the transition and integration period.

TABLE 2.1

DE&S Independent Assessment Team Membership

Maine Yankee SBLOCA Analysis Assessment

B. M. Dunn	Framatome Technologies, Inc.	Lynchburg, VA.
L. E. Hochreiter	Pennsylvania State University	State College, PA
G. B. Swindlehurst	Duke Energy Corporation	Charlotte, NC
Z. L. Taylor	Duke Energy Corporation	Charlotte, NC

Personnel Behavior Assessment

J. M. McGarry	Winston & Strawn	Washington, DC
M. J. Wetterhahn	Winston & Strawn	Washington, DC

Root Cause and Corrective Action Validation

D. C. Prevatte	Powerdyne Corporation	Fogelsville, PA

Engineering Process Assessment

R. W. Bonisolli	Duke Engineering & Services, Inc.	Bolton, MA
R. G. Eble	Duke Engineering & Services, Inc.	Vienna, VA
R. C. Futrell	Duke Engineering & Services, Inc.	Charlotte, NC
J. B. Stringer	Duke Engineering & Services, Inc.	Charlotte, NC
T. F. Wyke	Independent Consultant to DE&S	Charlotte, NC

Quality Assurance Assessment

J. Gerson	Duke Engineering & Services, Inc.	Richland, WA
P. R. Horsman	Duke Engineering & Services, Inc.	Charlotte, NC
D. A. Walker	Duke Engineering & Services, Inc.	Norcross, GA

Vermont Yankee Safety Analysis Assessment

J. Atchison	Scientech, Inc.	Idaho Falls, ID
B. Gitnick	Scientech, Inc.	Rockville, MD
D. Prelewicz	Scientech, Inc.	Rockville, MD
R. Tedesco	Scientech, Inc.	Rockville, MD

TABLE 2.1
(Continued)

DE&S Independent Assessment Team Membership

Seabrook Safety Analysis Assessment

W. Arcieri	Scientech, Inc.	Rockville, MD
T. Hencey	Scientech, Inc.	Clearwater, FL
D. Palmrose	Scientech, Inc.	Rockville, MD
D. Vreeland	Independent Consultant to Scientech, Inc.	

3.0 DE&S RESPONSE TO NRC CONCERNS

The NRC identified concerns in Section IV of the Demand that relate to questions regarding the efficacy of permitting DE&S to continue providing safety-related analyses to power reactor licensees. This Section provides the DE&S response to each of the stated NRC concerns.

3.1 Adherence to NRC Requirements

Statement of NRC Concern:

"The foregoing situation [as described within the Demand] raises serious questions concerning regard for and adherence to NRC requirements ....."

DE&S Response:

DE&S is 100% committed to providing products and services to NRC power reactor licensees that are unquestionably compliant with all applicable NRC requirements and the licensing basis of the nuclear station for which the work is provided. This commitment is part of the history, the experience, and the reputation of both Duke and DE&S.

Independent assessments by DE&S of the acquired YAEC organizations show that meaningful improvements have been made in the treatment of NRC requirements within workplace procedures. Training has been provided to affected personnel on NRC requirements. Additionally, current work products, primarily calculations and analyses are compliant with NRC methodology guidelines and exhibit knowledge of relevant NRC requirements. A strong bias to establish safety margins through use of conservative assumptions is present.

However, the DE&S evaluation also identified areas where improvements are warranted, primarily in strengthening executive leadership and strengthening licensing support of and interaction with ongoing work. Specific actions are underway to address these two areas. Effective March 1, 1998, Mr. William H. Rasin will become the DE&S Vice-President of Nuclear, Fuel, and Quality Assurance Services. This organization encompasses what were formerly the YAEC Nuclear Engineering, Fuel Management, and Quality Assurance Departments. [It should be noted that the LOCA Group resided within the YAEC Nuclear Engineering Department.] Appendix J provides a copy of Mr. Rasin's resume. The Nuclear Engineering Department has also been restructured and new management added to the department. Additionally, licensing functions associated with the acquired YAEC organizations will be fully integrated with existing DE&S licensing functions to strengthen the level of regulatory support provided to the workforce.

To provide additional assurance of quality work products that are in accordance with DE&S management expectations and in compliance with applicable NRC requirements throughout the transition and integration of the acquired organizations into DE&S, additional emphasis will be placed on the DE&S independent assessment process during the transition and integration period.

DE&S firmly believes that the actions described above, when considered in combination with other actions DE&S is undertaking and actions previously implemented by YAEC, assure that products and services provided to NRC power reactor licensees by DE&S comply with all applicable NRC requirements and the licensing basis of the nuclear station for which the work is provided.

3.2 Management Control

Statement of NRC Concern:

"The foregoing situation [as described within the Demand] raises serious questions ..... concerning management control and supervision over [NRC] licensed activities ....."

DE&S Response:

DE&S is 100% committed to maintaining effective management control and supervision of its safety related and NRC licensed activities. DE&S will not compromise on safety. A primary method of assuring compliance with this DE&S commitment is to conduct work, especially those related to nuclear safety, in a controlled and disciplined manner. This is also the culture Duke maintains to ensure successful execution of the responsibility to safely operate its three nuclear stations.

Independent assessments by DE&S of the acquired YAEC organizations show that a strong proceduralized work process that reflects NRC guidelines is in place. Significant improvements have been implemented in the deficiency reporting system and its active utilization is indicative of a workforce that openly identifies potential deficiencies. Additionally, the technical quality of work products demonstrates the effectiveness of the existing work processes in producing high technical quality. All are evidence of good management control.

However, the DE&S evaluation also identified areas where improvements are warranted, primarily in strengthening the workforce understanding of organizational roles and responsibilities and independent oversight. Specific actions underway to address these two areas include (1) clearly defining and formally documenting regulatory and organizational interface requirements with nuclear clients formerly supported by YAEC and (2) the DE&S independent oversight function as described in Section 3.1. Recognizing that organizational relationships are changing both internally and with nuclear clients formerly supported by YAEC, the first action will prevent recurrence of the communication and organizational responsibility uncertainties that played such a large role in the events described within the Demand. The independent assessment function will provide DE&S management with direct feedback on the quality of work products, their conformance with DE&S management expectations, and their compliance with applicable NRC requirements throughout the transition and integration of the acquired YAEC organizations into DE&S.

Other actions being undertaken by DE&S provide further assurances of effective management control. Among these are the management actions described above in Section 3.1, the activities to fully integrate the acquired YAEC work processes and organizations into DE&S described

throughout Section 2.3, and the management focus on strengthening the quality of workforce training as described in Subsection 2.3.2.

DE&S believes that actions previously implemented by YAEC have produced a good infrastructure. Additional DE&S management actions to integrate this infrastructure into DE&S to clearly define organizational responsibilities, emphasize strong technical oversight in the workplace, and strengthen technical leadership provide assurance that DE&S is exerting effective management control over all safety-related and potentially safety-related activities.

3.3 Accurate and Complete Information

Statement of NRC Concern:

"Questions are raised as to whether YAEC and/or DE&S will in the future provide complete and accurate information to licensees and to the NRC ....."

DE&S Response:

DE&S is 100% committed to providing accurate and complete information to its power reactor licensee clients and to the NRC. The commitment of DE&S to safety and the success of DE&S as a business mandate that all products and services provided to NRC power reactor licensees be accurate and complete. Through its affiliation with Duke, DE&S understands very clearly the importance of accurate and complete documentation in supporting nuclear station operations.

Independent assessments by DE&S of the acquired YAEC organizations show that the work products produced are consistently of a high technical quality, are consistent with NRC guidelines, and meet requirements. However, the assessment results also showed a variability in the quality of the supporting documentation. The guideline used to judge the adequacy of analyses, calculations and other technical reports and documents was whether the documents were sufficiently detailed as to purpose, method, assumptions, inputs, references, and units that a technically qualified individual could understand the documents and verify their adequacy without recourse to the originator. The assessments found some documentation which required a technical document's originator to provide clarifications or respond to review questions. Therefore, utilizing the guideline defined above indicated that more focus is required to enhance work product documentation quality. Within the scope of the DE&S assessment teams, no clear trends were identified other than a noticeable improvement in quality from older to more recent documents.

DE&S corrective actions are initially focused on conducting additional document reviews to determine whether any necessary programmatic actions are warranted. DE&S has initiated a number of joint client-DE&S technical teams to review client specific documentation for adequacy in meeting their business and regulatory needs. The quality of recent technical documents does show that current work practices and procedures are producing quality documentation. The independent assessment function described previously will provide DE&S management with direct feedback regarding the quality of current work product documentation. Additionally, DE&S actions underway to document roles, responsibilities, and

communication requirements with nuclear clients formerly supported by YAEC will strengthen the understanding of their documentation expectations and requirements. These client expectations and requirements will be communicated to the acquired organizations as part of DE&S transition training activities.

The DE&S project planning process requires clear definition and documentation of client requirements, methods used to ensure effective communication with the client, and training of project personnel on the requirements and provisions of the project plan. The DE&S project planning process is described in Appendix A, Section A.3.3. As described in Section 2.3, DE&S is transitioning ongoing work performed by the acquired YAEC organizations to this process. Completion of this transition will provide assurances of clear client communications at the individual project level.

DE&S believes that quality and accuracy of recent technical documents shows that the acquired YAEC work practices and procedures are producing quality documentation. However, older documentation may require recourse to the document's originator to provide clarifications or respond to review questions. DE&S believes that accurate and complete communication is currently being provided. Actions have been initiated with clients formerly supported by YAEC to ensure that each client's documentation needs and expectations are clearly understood and that existing documentation meets their business and regulatory needs. These actions, combined with the transition of work to the DE&S planning process and the definition of organization responsibilities described earlier, ensure that accurate and complete documentation is provided to licensees and the NRC.

3.4 Conduct of Work

Statement of NRC Concern:

"Questions are raised as to whether YAEC and/or DE&S are willing and able to conduct their activities in accordance with the Commissions requirements ..... "

DE&S Response:

DE&S is 100% committed to conducting its work activities in a manner that complies fully with all applicable NRC requirements and the licensing basis of the nuclear station for which the work is provided.

Independent assessments by DE&S of the acquired YAEC organizations show that: (1) a strong proceduralized work process that reflects NRC guidelines is in place, (2) the work products produced are consistently of a high technical quality, and (3) the work products produced are consistent with NRC guidelines and requirements. Additionally, the assessments noted that the workforce consistently demonstrates a high level of competence and knowledge within their technical areas of expertise, with an evident knowledge of industry practices and requirements.

However, the DE&S evaluation also identified areas where improvements were warranted. Actions DE&S has undertaken to strengthen work practices and formality of operations have

been described above. These actions include: (i) integration of the acquired YAEC work processes into the DE&S work processes, (ii) strengthening personnel training, (iii) formally documenting organizational roles, responsibilities, and communication requirements, and (iv) strengthening management leadership. Additionally, the independent assessment function described above provides DE&S management with direct feedback on the compliance of work process, practices, and products with DE&S management expectations and NRC requirements. Other actions underway that support work process improvements are described throughout Section 2.3

The basic DE&S work processes are summarized in Appendix A, Section A.3. DE&S firmly believes that using its existing work processes as a foundation, in combination with the actions described above and the actions previously implemented by YAEC, provide assurance that DE&S work practices and conduct of work activities comply with all applicable NRC requirements.

4.0 DE&S RESPONSE TO NRC ISSUES

The NRC has a specific set of technical issues described in Section III of the Demand that relate to YAEC's development and utilization of RELAP5YA, primarily with respect to SBLOCA analyses. This section provides a specific DE&S response to each of these issues.

4.1 LOCA Break Spectrum

Statement of NRC Issue:

During Cycle 14 operations, and in the Cycle 14 and Cycle 15 CPAR analyses, YAEC caused Maine Yankee to use apparently unacceptable evaluation models which could not calculate or reliably calculate ECCS performance. The models used were in apparent violation of 10CFR50.46(a)(1), because there was a region of the small-break spectrum between break sizes of 0.35 ft2 and at least 0.6 ft2 for which no acceptable evaluation model could either calculate or reliably calculate ECCS performance. The Manager and the Lead Engineer knew of this. In addition the oscillations and instability in the analysis became more severe at larger break sizes, increasing the risk that the limiting breaks had not been identified.

DE&S Response:

The Maine Yankee SBLOCA analysis performed by YAEC with the RELAP5YA code consisted of a spectrum of break sizes ranging from 0.1 ft^2 to 0.35 ft^2. The 0.35 ft^2 break size run was not completed prior to the PCT being reached. YAEC concluded that the limiting SBLOCA PCT was determined to be within this range of break sizes, and determined a limiting PCT value of 1887°F for the 0.15 ft^2 break. The rationale for this conclusion was documented as being based on the decreasing trend of PCT for smaller and larger break sizes. YAEC maintained that this scope of SBLOCA analyses met the 10CFR50 Appendix K and 10CFR50.46 regulations. In response to questions from the RELAP5YA SBLOCA Technical Issues Assessment Team (Review Team), the LOCA staff provided additional verbal justification consisting of an explanation of SBLOCA phenomena which supported their conclusion.

The prior licensing basis SBLOCA analysis performed by Combustion Engineering in 1977 established a trend of increasing PCT with break size, with the 0.5ft^2 break having a PCT of 1348°F. Other non-licensing basis analyses performed later by Siemens and ABB indicate both a local PCT peak near the limiting YAEC break size, and a trend of increasing PCT at the largest break size analyzed. Given the information provided and based on the trends of other analyses, the Review Team was unable to draw a definitive conclusion regarding the RELAP5YA PCTs for the unanalyzed portion of the Maine Yankee SBLOCA spectrum. However, it is concluded that the SBLOCA PCTs for all of these analyses meet the 10CFR50.46 2200°F criterion, and that SBLOCAs remain bounded by LBLOCAs.

DE&S agrees that the industry standard practice is consistent with the NRC's position that an Appendix K SBLOCA evaluation model must be capable of analyzing any break size within the licensed break range. This does not mean that all break sizes need to be analyzed, but rather that the model must be capable of analyzing them. The RELAP5YA evaluation model has not

demonstrated the capability to analyze the historical Maine Yankee SBLOCA break spectrum. To meet the expectations of the NRC, sound engineering arguments can be used but should be communicated to the NRC and agreed upon prior to implementation.

DE&S also agrees with the Review Team's concurrence with the NRC's position that the YAEC SBLOCA model produced oscillatory and unstable results. This behavior is evident following accumulator injection and in particular for the larger break sizes. These code problems were long-standing and widely known within the YAEC organization. The model is also considered to be unreliable in that an unexpected large change in results can occur with only a small change in the input to the code. YAEC considered the oscillations and instability to be non-physical, and that the PCTs predicted were sufficient. DE&S agrees with the Review Team position that the oscillations and instability are, in part, non-physical and due to fundamental limitations in the RELAP5YA code. Based on a broader knowledge of SBLOCA phenomena and results from other codes, YAEC was confident that SBLOCA was bounded by LBLOCA. Based on this expectation, YAEC accepted the results from RELAP5YA as adequate for showing compliance with the regulations. This may be a correct conclusion. However, YAEC's conclusion is based, in part, on information beyond the demonstrated results of runs of the RELAP5YA code for Maine Yankee. DE&S concludes that this situation should have been communicated to the NRC prior to implementation.

The YAEC report YAEC-1868, which summarizes the results of the application of the SBLOCA evaluation model to Maine Yankee, discloses the results of the analyses, including the break spectrum analyzed, the termination of the 0.35 ft^2 break case, and YAEC's explanation as to why the results meet the regulations. The illustrated results show the unstable and oscillatory behavior of the code. This report was reviewed by YAEC senior management and approved. YAEC-1868 was not submitted for NRC review based on YAEC's and Maine Yankee's understanding of a communication from the NRC Project Manager for Maine Yankee. Maine Yankee and YAEC expected that NRC review would be by a future inspection. Not submitting YAEC-1868 for NRC review was an error. YAEC and Maine Yankee should have understood the NRC's expectations that such a review was necessary prior to implementation.

As a consequence of the breakdown in the LOCA licensing process which resulted in implementation of the SBLOCA analyses for Maine Yankee Cycle 14 prior to obtaining NRC review and approval, YAEC did provide Core Performance Analysis Reports (CPARs) to Maine Yankee with the deficiencies described above. This situation was not a result of any deliberate action to avoid compliance with the regulations, but rather a failure to understand NRC expectations. DE&S does not consider this failure to have resulted in a reduction in the safety of the plant, since LBLOCA is limiting and sets the core operating limits. Had the SBLOCA analyses been completed it is likely that the core operating limits for Cycle 14 would have remained the same.

4.2 Materially Accurate and Complete Documentation

Statement of NRC Concern:

As a result of YAEC's preparation and review of YAEC-1868, YAEC provided MYAPCo with information that was not complete and accurate in all material respects, and thus caused MYAPCO to be in apparent violation 10CFR50.9(a). CPARs maintained for information and submitted to the NRC by MYAPCo, in support of Cycle 14 and Cycle 15 reload applications were apparently not complete and accurate in all material respects. MY relied on YAEC-1868 to prepare Cycle 14 and Cycle 15 CPARs in order to demonstrate compliance with 10CFR50.46. YAEC-1868, in its entirety, conceals the lack of an acceptable evaluation model to calculate ECCS performance for a portion of the break spectrum between 0.35 ft^2 and at least 0.6 ft^2. As a result of the OI investigation it appears that no Maine Yankee personnel realized that the RELAP5YA code failed at 0.35 ft2 or that there might be a portion of the break spectrum for which there was no acceptable evaluation model to calculate ECCS performance, and that no one at YAEC informed MYAPCo personnel that RELAP5YA nad failed at 0.35 ft^2.

DE&S Response:

YAEC-1868 documents the results of the SBLOCA analyses performed by YAEC using the NRC-approved YAEC-1300P SBLOCA evaluation model. DE&S agrees with the Review Team that this document was sufficiently complete and accurate, and an appropriate summary of the SBLOCA calculations performed. However, the downcomer modeling changes that were not discussed in YAEC-1868 should have been communicated to the NRC in YAEC-1868, as a revision to YAEC-1300P, or in some other communication. It is clear that YAEC did not appreciate the licensing significance of the downcomer modeling changes. YAEC-1868 was understandable to its intended audience, and was suitable for a licensing submittal in support of Maine Yankee. It is noted that the abstract is potentially misleading in the use of the word "complete", but that the scope of the analysis as contained within the report is characterized correctly. The amount of technical information included was appropriate for any knowledgeable person to understand the results of the analysis. There was no intentional concealment of information which would have identified any non-compliance with NRC regulations. The statements in the report regarding compliance with the NRC's regulations were consistent with the YAEC LOCA Group's understanding of the regulations. DE&S concludes that YAEC's understanding of the regulations was not consistent with the understanding of peers in the industry LOCA community. The compliance statements and the supporting analyses in YAEC-1868, as interpreted by a knowledgeable engineer not trained in the LOCA licensing process, could be understood as a logical basis for compliance. Therefore, YAEC-1868 could be understood by a knowledgeable engineer not trained in the in LOCA licensing processes to be complete and in compliance with the regulations.

Based on discussions with YAEC personnel, the problems with applying the RELAP5YA code to Maine Yankee were a regular topic of discussion over several years with cognizant Maine Yankee personnel. Although utility staff are generally not LOCA experts, people assigned to interface with LOCA organizations generally have sufficient knowledge to understand the subject, review documents, and make appropriate decisions on behalf of the utility. However,

utility staff generally rely on the LOCA organization to manage the technical details associated with compliance with Appendix K and 10CFR50.46. DE&S agrees with the Review Team that Maine Yankee staff, similar to typical utility staff, should have relied on the Yankee LOCA organization for LOCA licensing support. It was YAEC's understanding that Maine Yankee directed YAEC not to independently interface with the NRC staff. This was a root cause in the subsequent licensing process failures. Had YAEC executive management been more aggressive in maintaining the licensing function, YAEC personnel would have been better prepared to make decisions relative to NRC expectation and the interface with the NRC. This situation resulted in an absence of communication with the NRC LOCA technical staff for a number of years.

4.3 Evaluation Model Conservatism

Statement of NRC Issue:

During Cycle 14 operations and in the Cycle 14 and Cycle 15 CPAR, YAEC caused MYAPCo to use an apparently unacceptable SBLOCA evaluation model which over predicted core cooling. YAEC-1868 apparently did not satisfy the requirements of 10CFR50.46(a)(1), because as a result of incorrect calculations of the penetration factors, which arose from misapplication of the Alb-Chambre penetration correlation, the analysis provides no basis to assume full penetration of the emergency core cooling system injection and provides no basis to derive the loss coefficient of 600 used for the split downcomer nodalization. These deficiencies resulted in over prediction of core cooling and overstatement of the conservatism of the model. If the Alb-Chambre correlation had been applied correctly, penetration factors would have been calculated in the range of -0.6657 to -0.7767, which is a meaningless result because the calculations would have been less than zero. Such calculations also indicate other possible errors in application of the Alb-Chambre correlation. Adequate QA review would have revealed the errors and the unacceptability of the RELAP5YA SBLOCA analysis described in YAEC-1868.

DE&S Response:

Early applications of the RELAP5YA SBLOCA evaluation model to Maine Yankee identified excessive ECCS bypass relative to what was expected based on scaled test facility data and the results of other codes. Revisions to the interfacial drag model were only partially successful in addressing this non-physical prediction. Various modeling approaches were attempted to make the ECCS penetration into the vessel lower plenum more physical. Eventually an artificially large loss coefficient was introduced in the junction connecting the two volumes representing a split reactor vessel lower downcomer. The value of this loss coefficient was varied to obtain a balance between the expected ECCS penetration and the effect on steam venting via the break. A value of 600 was selected as an appropriate value. The amount of ECCS penetration obtained with this modeling approach was based, in part, on the Alb-Chambre correlation. This correlation was applied to confirm that the amount of ECCS penetration predicted by RELAP5YA was conservative. DE&S agrees with the Review Team that this modeling approach was justified, and the use of a value of 600 obtained an amount of ECCS penetration that was consistent with industry experience. This modeling approach is not expected to result

in an overprediction of core cooling, but since the calculations were not completed for all break sizes, it cannot be definitively confirmed.

An error was made in the application of the Alb-Chambre correlation. This arithmetic error was not identified during the quality assurance process. This is a failure of the quality assurance process. Closer investigation of this failure offers a reasonable explanation as to why it occurred. The arithmetic error did not skew the result of the calculation (value of 8), which was in the range of the expected result that complete penetration was predicted. The correlation can produce results in excess of the value of 1, which have the meaning of complete ECCS penetration. A person performing a quality assurance review is influenced by the result based on experience and expectations. These are the most difficult errors to identify. A correct application of the Alb-Chambre correlation without the arithmetic error would have produced negative values indicating complete ECCS bypass. This result would have been immediately recognized by YAEC as non-physical for the SBLOCA conditions of interest. The cause of the non-physical result would have been traced to excessively conservative input values. More reasonable values would then be input to the correlation, and reasonable and valid results indicating significant ECCS penetration would have resulted. Therefore, although the application of the Alb-Chambre correlation was in error, the modeling which incorporated the loss coefficient with a value of 600 remains valid. Thus the results of the error in applying the Alb-Chambre correlation did not result in invalid input to the SBLOCA analyses.

DE&S agrees with the Review Team that use of the Alb-Chambre correlation as the basis for the modeling approach which includes the junction loss coefficient of 600 in the reactor vessel downcomer is reasonable. It is recognized that the PCT results of the SBLOCA analyses for some of the break sizes are very sensitive to the value of the loss coefficient. This entire modeling approach was not presented to the NRC for review. YAEC considers the value of the downcomer junction loss coefficient to be an input to the evaluation model. This is literally true, since all junction loss coefficients and most of the plant applications model are in the form of inputs. However, the Review Team concludes that due to the non-physical value used and the significance of this input parameter, it is in reality more of a model change, and in fact a very important aspect of the SBLOCA evaluation model. DE&S agrees with the Review Team's conclusion. This model should have been submitted to the NRC for review. It is also recognized that had there been a submittal of the Maine Yankee SBLOCA applications to the NRC, this modeling approach would have been discussed and would have been reviewed. DE&S supports the Review Team's position that this model could have been approved by the NRC in this form or with some revision.

4.4 Best Estimate Analysis Utilization

Statement of NRC Issue:

YAEC caused Maine Yankee to apparently violate 10CFR50.46(a)(1) by relying on an unacceptable SBLOCA evaluation model (Best Estimate RELAP5YA SBLOCA evaluation model) to calculate ECCS cooling performance in preparing a Section 50.59 analysis used to determine if a decrease in steam generator pressure involved an unreviewed safety question. Additionally, the proposed BE RELAP5YA evaluation model was not the approved SER computer code and did not comply with 10CFR50, Appendix K requirements. The NRC

indicates that is also reasonable to conclude that the Manager knew that the analysis which YAEC performed regarding the effects of a reduction in steam generator pressure on LOCA analyses was a safety analysis which would be used by Maine Yankee in a Section 50.59 analysis or other safety analysis. In view of the intended use of the YAEC analysis, the Manager should have provided Maine Yankee with an analysis which met NRC requirements.

DE&S Response:

Due to difficulties in applying the YAEC-1300P Appendix K RELAP5YA SBLOCA code to the Maine Yankee plant, the YAEC LOCA Group initiated a parallel effort to develop a "best-estimate" (BE) LOCA evaluation model, and so indicated in a memorandum. This modeling approach is an alternative approach to the traditional Appendix K approach. Since the YAEC-1300P topical report did not include this BE approach, a separate NRC approval of this method would be necessary prior to implementation. The Yankee LOCA Group Manager clearly understood that the BE approach required further NRC review. The possibility of using the proposed BE approach to satisfy NUREG-0737 Item II.K.3.31 for Maine Yankee had been discussed with the Maine Yankee staff. A report describing the proposed BE methodology was sent to the Maine Yankee staff. The YAEC LOCA Manager was under the impression that the report had been submitted to the NRC for review.

In 1990 Maine Yankee initiated a service request with YAEC to perform a scoping analysis of the effect of reduced steam generator pressure on the licensing basis transients and accidents. Steam generator pressure was decreasing steadily due to fouling of the steam generator tubes. The YAEC LOCA Group was responsible for addressing the LOCA aspects of this issue. In order to be responsive to the customer's request, YAEC used the only available LOCA analysis tool at that time, the BE model, to assess the effect of reduced steam generator pressure on the analysis of record. Memorandum LOCA 91-04 dated January 25, 1991 documented the use of the BE SBLOCA model. In this memorandum the BE model is erroneously stated to be the "licensing basis SBLOCA analysis". The LOCA Group Manager was apparently out of the office when this memorandum was signed off, and the Department Manager signed off on the approval. It is apparent that the LOCA Group staff was confused about what the actual SBLOCA licensing basis was at this time. The analysis of record at this time was the 1977 Combustion Engineering analysis.

The analysis documented in LOCA 91-04 was eventually transmitted by reference to Maine Yankee as part of two other memoranda (NED 91-18 dated January 28, 1991, and TAG-MY-92-035 dated May 29, 1992). The LOCA Group Manager did sign his approval on the TAG-MY-92-035 memorandum. It is noted that the SBLOCA discussion in this memorandum is a very small part of the technical content and there is no mention of the referenced analysis as using the BE model. In April 1992, YAEC noted on a service request form (92-42) related to the reduced steam generator pressure that the Appendix K SBLOCA analyses were nearing completion and that YAEC would provide these results to Maine Yankee when completed. However, the BE SBLOCA analysis memorandum was referenced by Maine Yankee in a 50.59 evaluation dated April 12, 1993 as part of the justification for operation with reduced steam generator pressure. The Appendix K SBLOCA results were forwarded to Maine Yankee on April 12, 1993, along with a draft 50.59 for Maine Yankee's

use. These results were not referenced by Maine Yankee until January 13, 1994, when the 50.59 was revised.

In the various memoranda sent to Maine Yankee by YAEC that applied the BE SBLOCA model there were never any statements to the effect that the model was not NRC-approved. The LOCA 91-04 memorandum actually stated the contrary, which was incorrect. This indicates a lack of administrative control and communication within the Yankee LOCA Group and within the YAEC Maine Yankee Projects Group. It also indicates a failure to appreciate the disclosures that should be made regarding application of unlicensed LOCA analysis technology in responding to plant support requests. There was also a failure to correct the situation when the results of unlicensed methods were found to have been released to the customer.

DE&S disagrees with the NRC position that unlicensed LOCA analyses cannot be used in performing some scoping safety evaluations, including input to 50.59 evaluations. Such applications can be appropriate provided that the application does not replace the analysis of record, and provided that the use of the unlicensed analysis method is clearly stated and justified. If there is any doubt regarding the appropriateness of such an application, then the NRC should be consulted prior to implementing the results of the analysis. YAEC's failure in this situation was in incorrectly characterizing the BE SBLOCA analysis as the licensing basis, and furthermore not stating that the analysis used unlicensed methods including restrictions on its use. In providing any licensee with a document, the supplier must consider possible uses of the document. If the use of the document must be restricted, such as one which uses unlicensed methods, the document must clearly state so.

It should be noted that the effect of reduced steam generator pressure on the SBLOCA results for the magnitude of steam generator tube fouling and plugging that was being evaluated would be expected to not be significant. This is particularly true given that the analyses of record showed the SBLOCA PCTs to be significantly lower than the LBLOCA PCTs. An evaluation could have been justified without any SBLOCA analysis. An evaluation could also have been justified using BE analysis methodology provided that sufficient qualification was provided, and provided that the analysis of record was not replaced. Since such qualification was not provided with this evaluation which included references to the application of the BE SBLOCA methodology, this approach was not justified.

5.0 DE&S RESPONSE TO NRC DEMAND FOR INFORMATION

Section V of the Demand requested DE&S to provide information with respect to two topics. Section 5.1 of this report provides the DE&S response to the NRC question regarding continued performance of safety-related analyses for NRC Licensees. Section 5.2 of this report provides the DE&S response to the NRC question regarding the willfulness of DE&S personnel behavior with respect to the actions described within the Demand.

5.1 Continued Performance of Safety-Related Analyses

Statement of NRC Information Need:

"[Provide] an explanation why, in view of the matters set forth above [in the Demand], the NRC should permit any NRC Licensee to use the services of YAEC LOCA Group and/or DE&S, to the extent that YAEC LOCA Group was transferred to DE&S, to perform LOCA analyses or any safety-related analysis to meet NRC requirements."

DE&S Response:

DE&S is committed to the highest standards of integrity and technical quality upon which its clients and the NRC can rely upon without reservation. As a wholly-owned affiliate of Duke, DE&S carries forward to the nuclear industry Duke's nuclear tradition built through four decades of nuclear design, construction, licensing, and operations experience. This tradition not only encompasses strict attention to and understanding of regulatory and industry requirements, but also emphasizes a technically strong workforce who are held accountable for their performance. The principles of: (1) adherence to NRC requirements, (2) effective management control of safety-related activities, (3) accurate and complete communication with licensees and the NRC, and (4) conduct of work in accordance with NRC requirements are fundamental to the Duke/DE&S nuclear tradition. As a provider of management and technical services to the nuclear industry, DE&S cannot succeed as a commercial business without the total confidence of its clients and the NRC in this DE&S commitment to integrity and quality.

DE&S has completed an extensive set of independent assessments of the acquired YAEC organizations. These assessments focused on ensuring that the acquired YAEC organizations are currently providing quality technical products and services to their NRC licensee clients, in a manner that is compliant with NRC requirements. In fact, many of these same clients conducted licensee oversight activities during these assessments. The DE&S assessments addressed both process and technical aspects of the acquired YAEC organizations. Significant focus was placed on LOCA and associated safety analyses, as well as engineering work practices and products, quality assurance activities, and the effectiveness of corrective actions implemented by YAEC. The specific DE&S assessments conducted were:

1. Maine Yankee SBLOCA Analysis Assessment - an independent technical review of the technical issues described in the Demand.

2. Personnel Behavior Assessment - an independent legal review of the "willfulness" of the actions and events summarized in the Demand.

3. Root Cause and Corrective Action Assessment - an independent review of the YAEC root cause assessment and corrective actions.

4. Engineering Process Assessment - an independent review of engineering/technical work practices used and work products produced during the past 3 years, other than those associated with LOCA analyses.

5. Quality Assurance Assessment - an independent review of QA audits performed during the past 3 years.

6. Vermont Yankee Safety Analysis Assessment - an independent technical review of: (i) Basis for Maintaining Operation (BMO) documents, (ii) Cycle 20 reload analyses, and (iii) containment analyses.

7. Seabrook Safety Analysis Assessment - an independent technical review of radiological consequence, non-LOCA transient analyses.

DE&S also assessed the underlying causes identified by YAEC that led to the events described in the Demand. These are:

1. Definition of organizational roles, responsibilities and communication requirements between YAEC and MYAPCo was less than adequate,

2. Training of YAEC personnel was less than adequate, and

3. Clarity and specificity of procedures was less than adequate.

Descriptions of the DE&S assessments, conclusions drawn from these assessments and follow-up actions identified by DE&S are provided in Sections 2.2. and 2.3. More detailed descriptions are provided in Appendices C - I.

DE&S management evaluated the assessment results to ensure that products and services being provided are high quality and meet the needs of DE&S clients and the NRC. Additionally, areas for DE&S management attention were identified to ensure that products and services provided by the acquired organizations continue to meet the standards and expectations of DE&S management, the NRC and DE&S clients throughout the transition and integration of the acquired YAEC organizations into DE&S.

DE&S has concluded that the acquired YAEC organizations are providing quality products and services. In fact, DE&S did not identify any areas where the current performance of, or the products and services provided by, the acquired YAEC organizations is unacceptable. DE&S has also noted that meaningful improvements have been made in the quality of YAEC procedures and the emphasis these procedures place on NRC requirements and NRC reporting. Additionally, DE&S has consistently found that the technical quality of the work products currently being produced, as well as the professionalism and technical competence of the workforce, are high. In summary, the DE&S assessments have shown that the acquired

organizations fundamentally are technically sound, producing quality products and services in accordance with NRC requirements and industry practices.

The DE&S assessments and management evaluation did identify areas that warrant DE&S management focus to ensure a successful transition and integration of the acquired YAEC organizations into DE&S. Although a number of work activities are required to effectively transition and integrate an acquired organization into DE&S, the transition assessments generally highlight a few items warranting management attention to ensure a successful transition and integration. In the case of YAEC, four items were identified. These four items are: (1) organizational roles, responsibilities, and communications, (2) personnel training, (3) nuclear regulation support, and (4) accurate and complete documentation. DE&S follow-up actions that address these four items are described in Section 2.3. Each of these actions is either completed or underway. These actions are summarized as:

(1) Jointly defining and documenting with each nuclear client formerly served by YAEC the organizational roles, responsibilities, expectations, and communication requirements between DE&S and the client.

(2) Transitioning ongoing work performed by the acquired YAEC organizations into the DE&S project planning process. This process includes identifying requirements for clear definition and documentation of project level technical and organizational communication requirements, and training of project personnel to these requirements.

(3) Strengthening the executive leadership in the Bolton office by naming, effective March 1, 1998, William H. Rasin as DE&S Vice-President of Nuclear, Fuel, and Quality Assurance Services. This organization includes the former YAEC Nuclear Engineering, Fuel Management, and Quality Assurance Departments.

(4) Communicating formally to the acquired YAEC organizations DE&S management expectations on: (1) the role and responsibilities of DE&S as a commercial provider on nuclear services and products to NRC Licensee clients, (2) the requirement to ensure that products and services meet the needs of the clients, and (3) the requirement to ensure that supporting documentation files are complete and accurate.

(5) Strengthening employee training through prompt integration of the acquired YAEC training functions into DE&S training functions, supporting the prompt identification of training needs and implementation of a systematic program for meeting these needs.

(6) Jointly reviewing with DE&S clients their client-specific documentation to ensure that supporting documentation meets each client's unique business and regulatory needs.

(7) Strengthening nuclear regulatory support to ongoing work through the prompt integration of DE&S licensing functions into the acquired YAEC organizations in such a manner that appropriate licensing support is provided.

In summary, DE&S found through its assessment process that the performance of, and the products and services provided by, the acquired YAEC organizations are acceptable.

Meaningful improvements have been made in the quality of YAEC procedures and the emphasis these procedures place on NRC requirements and NRC reporting. The technical quality of the work products being produced, as well as the professionalism and technical competence of the workforce, are high. DE&S is systematically transitioning the acquired YAEC organizations to the DE&S project planning and work execution programs and procedures. The DE&S project planning process emphasizes clear definition and documentation of project-specific requirements, both technical and organizational, and training of project personnel on these requirements. DE&S has strengthened the Bolton office leadership with a proven nuclear industry executive, William H. Rasin. DE&S is also currently engaged with each nuclear client formerly supported by YAEC to ensure that roles, responsibilities, and expectations are mutually understood and documented. For these reasons, combined with the nuclear commitment and experience represented by DE&S, the NRC should have a high degree of confidence that safety-related analyses, products, and services provided by DE&S to NRC power reactor licensees comply with the principles of: (1) adherence to NRC requirements, (2) effective management control of safety-related activities, (3) accurate and complete communication with licensees and the NRC, and (4) conduct of work in accordance with NRC requirements

5.2 Willfulness of Personnel Behavior

Statement of NRC Information Need:

"[Provide] an explanation why the NRC should not consider the inadequate analyses, which apparently caused MYAPCO [Maine Yankee Atomic Power Company] to be in violation of NRC requirements, to be the result of willful, either deliberateness or careless disregard, on the part of YAEC and/or DE&S personnel."

DE&S Response:

A common finding or observation of the DE&S assessment teams was the high degree of professionalism and openness of the DE&S employees within the acquired YAEC organizations. DE&S has found no evidence that would question the sincerity and dedication of this workforce. The Demand focused on the actions of two individuals in relation to whether willfulness, either deliberateness or careless disregard, existed. DE&S found no willfulness on the part of the two individuals. DE&S, therefore, believes that the Manager and Lead Engineer are capable of conducting their activities in the future in conformance with NRC requirements. Additionally, DE&S concludes that there was nothing found by the independent assessment of the conduct of YAEC (or DE&S) or its personnel that would prevent DE&S activities from being performed in full compliance with NRC requirements and expectations. Appendix D provides detailed assessment of the willfulness associated with the events described within the Demand.

Willfulness, as used by the NRC, embraces a spectrum of actions ranging from deliberate intent to violate or falsify to and including careless disregard for NRC requirements. Deliberate misconduct is defined as "an intentional act or omission that the person knows would cause a licensee to be in violation of any regulation or other NRC requirement." On the other hand, a finding of careless disregard indicates that the person acted with reckless indifference to a

requirement or with disregard or utter unconcern of the consequences of whether there was compliance. The existence of a reasonable justification for an action would defeat a charge of willfulness despite the fact that a person undertook an action that was ultimately found to violate NRC requirements.

DE&S found no actions on the part of any individuals associated with the specific issues contained in the Demand that would constitute deliberateness. All individuals where found to be open, honest and communicative. No specific intent to violate any NRC regulation has been identified. The DE&S evaluation focused on whether there existed careless disregard for Commission requirements.

There were four specific allegations in the Demand, the first two of which had a common factual underpinning. The NRC alleged that because not all points of the SBLOCA spectrum could be reliably calculated by the code used by YAEC for the Maine Yankee facility, the requirements of 10CFR50.46 were not met. The RELAP5YA SBLOCA Technical Issues Assessment (Technical Review Team) concluded that the standard industry practice, as utilized by experts in the LOCA field, was that the code should have the capability of analyzing all points within the prescribed spectrum. YAEC had taken the position that the identification of the limiting break, combined with a sufficient understanding of the physical phenomena which were occurring over the entire small break region, provided compliance with 10CFR50.46. It is recognized that NRC expectations are not completely documented regarding 10CFR50.56, but rather had been communicated to the LOCA community through its interactions with NRC. Utilizing the appropriate legal standard, and bearing in mind that the YAEC LOCA Group was isolated and not part of the LOCA community, it was determined that the applicable regulation could be read as it was by YAEC. Actions to implement the LOCA Group's interpretation of the regulations did not evidence reckless indifference and, thus, no careless disregard was found.

With regard to the second issue, the NRC asserted that YAEC caused a violation of 10CFR50.9 in that it provided inadequate information to Maine Yankee regarding the SBLOCA which did not reveal the code inadequacies discussed above. Many of the same considerations apply to this issue as to the previous one. In addition, the Technical Review Team found that on the whole the document submitted to Maine Yankee was sufficiently complete and accurate when judged using the perspective of its intended audience, i.e., one knowledgeable in the field, such as an NRC reviewer. Under the circumstances, DE&S found that careless disregard of the regulations was not present.

The third issue relates to an assertion by the NRC that Maine Yankee had not provided a technical basis for one element of the SBLOCA analysis, the loss coefficient for the split downcomer nodalization, and, as a result, there was overprediction of core cooling and overstatement of the conservatism of the model. The Technical Review Team determined that the modeling approach utilized was reasonable and consistent with industry experience. The Technical Review Team determined that a deficiency in the quality assurance of a confirmatory calculation existed, but there was a reasonable explanation as to why it occurred, and that the evaluation undertaken by YAEC was appropriate. No inadequate analysis existed and the issue of deliberateness or careless disregard did not arise.

With regard to the fourth and last issue, the NRC asserted that YAEC memoranda used an unacceptable Best Estimate model rather than an approved Evaluation Model in evaluating the effect of a decrease in steam generator pressure on the peak clad temperature in the small break region when it should have known that such analysis would form the basis of a 10CFR50.59 analysis. The Technical Review Team determined that methods other than the approved Evaluation Model could have been utilized for the work undertaken. However, the Technical Review Team felt that limitations on the use of such methods should be stated. Here certain memoranda mischaracterized the Best Estimate model as the approved 10CFR50.46 Evaluation Model for the facility.

With regard to the use of the memoranda, DE&S determined that it was not unreasonable to have failed to contemplate that such work products would be used in a 50.59 analysis inasmuch as they either represented scoping calculations and/or were not thought at the time to be addressing design basis issues. However, it is also understandable that a Maine Yankee employee not expert in LOCA analyses could use these memoranda in performing the assigned 50.59 analyses. In evaluating whether careless disregard had occurred, it was noted that the issue of whether decreased steam generator pressure impacted the design basis did not mature until late 1992 (i.e., some months after receipt of the last memorandum), as evidenced by the Maine Yankee NRC Resident Inspector's suggestion that a 10CFR50.59 analysis should be performed. It was also noted that the appreciation of the necessity for evaluating degraded conditions pursuant to 10CFR50.59 was in a state of transition a the time, with the NRC publishing Generic Letter 91-18 in late 1991. DE&S concluded that while a misstatement was made that the Best Estimate model was the licensing basis SBLOCA analysis and while it was used without prior discussion with the NRC, the actions of YAEC personnel did not meet the test for careless disregard of the regulations in that DE&S could not conclude that there existed a reckless disregard or careless indifference towards its responsibilities or the consequences of the actions taken under the circumstances assumed. Indeed, on the very date Maine Yankee's 10CFR50.59 analysis was internally approved, YAEC furnished Maine Yankee a draft 50.59 analysis on the precise subject based on its RELAP5YA Appendix K Evaluation Model.

In conclusion, DE&S determined that while in certain instances there may have been inadequate analysis associated with the SBLOCA analysis, there was neither deliberateness nor careless disregard resulting from the deficiencies discussed in the Demand. DE&S concludes that there was nothing identified by the independent assessment of the conduct of YAEC (or DE&S) or its personnel that would prevent DE&S activities from being conducted in full compliance with NRC requirements and expectations.

APPENDIX A

DESCRIPTION OF DUKE ENGINEERING & SERVICES, INC.

APPENDIX A

DESCRIPTION OF DUKE ENGINEERING & SERVICES, INC.

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A.1 COMPANY OVERVIEW

A.1.1 Company History

Duke Energy Corporation (Duke) began offering management and technical services to electric utilities and related businesses in 1982. Five years later, Duke Engineering & Services (DE&S) was incorporated as a wholly owned affiliate of Duke to provide these services to clients. (Figure A-1 shows the relationship of DE&S within the Duke corporate structure.) DE&S has grown continuously and expanded both the range and scope of services offered. Consistent with these developments, DE&S has gained considerable experience in working under a variety of organizational structures, teams and implementation formats. Because Duke is the only U.S. utility performing all major elements of a commercial nuclear program with its own resources, DE&S is uniquely qualified to bring the full spectrum of commercial nuclear experience to clients. Through DE&S, Duke's four decades of nuclear design, construction, licensing, and operations expertise is made available to other clients worldwide. With one of the strongest backgrounds in the nuclear industry, DE&S offers clients a depth of engineering experience and expertise available through only a few companies in the world..

In 1996 DE&S expanded its resource base and scope of nuclear services available to clients with the acquisition of VECTRA Technologies, Inc.'s (VECTRA) nuclear engineering, power services and government services businesses. DE&S continued its commitment to the nuclear industry in 1997 with its acquisition of the Nuclear Services, and related support divisions, of YAEC. DE&S currently provides engineering, environmental, hydroelectric, nuclear, renewable energy, power transmission, oil/gas production, and oil/gas processing services to clients world wide. Figure A-2 provides the current DE&S corporate structure.

A.1.2 History of Nuclear Involvement

Nuclear power generation is a success story at Duke. From the experimental work of the Parr Station beginning in 1957, through the completion of Catawba Nuclear Station, Duke has performed and excelled in the complete process of licensing, procuring, designing, constructing, operating, and maintaining commercial nuclear power plants. With more than 20 years of large capacity nuclear power generation experience, Duke's nuclear plants are continually recognized for their outstanding records in operational effectiveness. Oconee Nuclear Station holds the record for total electric generation in the U.S. McGuire Nuclear Station has received the top fuel efficiency ranking for multi-fuel stations. Catawba Nuclear Station has received the best thermal efficiency rating among pressurized water reactors in the U.S.

Duke's history of nuclear involvement began with the Oconee Nuclear Station. Its three units were completed in just seven years at a cost of less than half the industry average at that time. Today, Oconee remains one of the most efficient and least expensive generating stations in the country. In July 1991, Oconee Unit 1 became the first U.S. nuclear unit to produce 100 billion kilowatt-hours. McGuire was the next nuclear station and has consistently been one of the most efficient nuclear plants in the nation. When completed in 1986, Catawba, like the other stations, showed construction costs significantly below

industry averages. This station also enjoys a reputation for efficiency in operations and maintenance.

In 1985, the Electric Power Research Institute (EPRI) began to develop plans for a standardized Advanced Light Water Reactor (ALWR) to meet the needs of utilities in the future. This program was led by the utilities in order to assure that the needs of the owner and operator were addressed in the next generation of nuclear plant designs. Based on extensive nuclear experience, Duke played a central role in the development of the EPRI ALWR Requirements Document and subsequent developments such as the First-of-a-Kind Engineering Program.

A.1.3 Industry Achievements

Year after year, Duke's stations rank among the top in the nation for efficiency. The company's nuclear program is widely recognized and respected. According to *Electric Light & Power*, all three Duke nuclear plants were ranked in the top twenty nuclear plants in 1993 and 1994, based on lowest total O&M costs. Duke also aims for each station to be in the top 25 percent of the industry in capacity factor. The industry's top quartile capacity factor in 1994 was 90 percent.

Recently, Duke was honored by being named the first three-time recipient of the electric utility's highest honor — the Edison Award. The award, named for inventor Thomas Edison, was also presented to Duke in 1972 and 1984.

Consistent with the company's goal of increasing nuclear plant safety and reliability, Duke played a leadership role in establishing two important industry exchange and oversight groups — the Institute of Nuclear Power Operators (INPO) and the World Association of Nuclear Operators (WANO). At WANO's inaugural meeting in Moscow in May 1989, Duke's former chairman, William S. Lee, was named first president, serving until 1991. In addition, Lee served as Chairman of The Board for INPO from its founding in 1979, until 1982.

Duke was one of the first utilities in the nation to establish its own training program for nuclear plant operators. Duke offers comprehensive instruction for nuclear plant operators as well as for technicians in health physics, chemistry, maintenance, instrumentation and electronics. In August 1983, the Oconee Nuclear facility became the first INPO accredited Plant Operator Training Program facility. All operator training programs are now accredited by INPO and have been approved by the NRC. Duke utilizes its on-site, full scope, control room simulators for initial license and requalification training of operators. All simulators are maintained with the highest degree of fidelity to the reference plant according to established standards for simulator configuration management.

A.1.4 Engineering Expertise

DE&S brings a wealth of engineering expertise to clients' projects in civil/structural, electrical/I&C and mechanical/nuclear disciplines. Since 1904, when Duke Power Company's first hydroelectric station was constructed, employees have performed siting,

quality assurance, design, licensing, construction, operations, maintenance and modifications for all its nuclear, fossil, hydroelectric, and pumped storage facilities. A complete multi-discipline staff is maintained to provide ongoing engineering support to station operations and maintenance personnel. This staff assists with the resolution of daily problems involving all systems and equipment, participates on evaluation teams, and provides engineering designs for the major plant modifications performed each year. Many DE&S employees began their careers with Duke.

DE&S/Duke employees represent all facets of engineering, including:

- Engineers with civil/structural engineering expertise who are experienced in all facets of civil/structural engineering including foundation and structural design, piping analysis, support/restraint design and environmental engineering.

- Engineers with mechanical engineering expertise and engineers with nuclear engineering expertise who are experienced in all facets of mechanical/nuclear engineering including mechanical systems and equipment design, HVAC, fire protection, piping, materials, nuclear fuels, safety analysis and radiation protection.

- Engineers with electrical/I&C engineering expertise with experience in all facets of electrical/I&C engineering including electrical systems and equipment design and I&C design.

Approximately 75 percent of DE&S engineers are registered professional engineers.

The DE&S northeast regional office, located in Bolton, Massachusetts, is comprised of a 500-member technical staff. This team includes nuclear, systems, mechanical and electrical engineers, licensed asbestos abatement supervisors, maintenance, welding and materials, nondestructive examination, instrumentation control, radiological and civil engineers; safety analysts; nuclear physicists; environmental, materials and metallurgical scientists; computer systems analysts; quality assurance, training and administrative support personnel. This staffing represents a consolidation of the existing DE&S northeast region staff with the acquired staffing of YAEC.

A.1.5 Regulatory/licensing Expertise

DE&S maintains a comprehensive licensing and regulatory compliance staff capable of dealing with the interwoven and complex regulations controlling nuclear and environmental work. Ranging from obtaining environmental permits at the local level, to supplementing the operational experience programs of a plant on the watch list, to licensing an ISFSI, DE&S licensing and regulatory compliance personnel and programs support nuclear clients every day. The YAEC acquisition compliments DE&S expertise with its direct experience in preparation for and participation in the NRC decommissioning hearing process, including interactions with stakeholders, based on the successful disposition of contentions raised in Yankee Rowe litigation.

In addition, DE&S experience includes development of licensing strategies for spent fuel storage under 10CFR Part 72 and Yankee Rowe's 10CFR Part 71 Certificate of Compliance application to the NRC and DOT for transportation of the Yankee Rowe steam generators as their own shipping casks.

DE&S assisted in the development of the Consumers Power Company's Big Rock Point Decommissioning Plan by providing both licensing and engineering services. DE&S advised Consumers Power on licensing strategies for addressing regulatory program requirements in a permanently shutdown mode and coordinated the development of the Decommissioning Plan and the accompanying Environmental Report. Additionally, DE&S supported licensing the recently completed steam generator replacement at St. Lucie 1.

DE&S also manages the effort of Louisiana Enrichment Services (LES) to license its Claiborne Enrichment Center, a proposed uranium enrichment facility to be located near Homer, Louisiana. This licensing effort is considered by many to be a bellwether for the future of commercial nuclear power and is being closely monitored by many inside and outside of the U.S. nuclear industry.

DE&S has, from the enactment of the National Environmental Policy Act (NEPA), addressed the many radiological and non-radiological aspects of licensing and operating nuclear generating facilities. This long history includes the preparation of Environmental Reports to assess the environmental impacts from proposed and operating facilities as well as facilities that may be allowed to operate beyond the original 40-year license term.

A.1.6 Nuclear Operations and Maintenance Expertise

As an affiliate of Duke Energy, DE&S offers nearly three decades of nuclear operations and maintenance experience, along with a unique blend of technical excellence, cost consciousness, regulatory adherence, environmental sensitivity and quality assurance. In 1995, Duke's three nuclear plants operated at a capacity factor of 90%, compared to the national average of 78%. Nuclear operations is a primary responsibility of DE&S as a member of the Management and Integrating Contractor teams at the U.S. DOE's Hanford Site and Idaho National Engineering Laboratory.

DE&S works closely with clients to shorten outages, enhance efficiency and upgrade plant output, as well as assist with operational readiness reviews, integrated outage management, engineering support, risk assessments, in-service testing, instrument calibration and system modifications.

DE&S' shift advisor/mentor program provides real-time assessment, technical advice and operating philosophy coaching to on-shift operations management. By identifying areas for improvement and implementing solutions, DE&S shift advisors and mentors have improved operating personnel performance and overall plant performance at a number of U.S. nuclear stations including Comanche Peak Steam Electric Station, Indian Point 3, Watts Bar Nuclear Plant and Salem Nuclear Generating Station.

DE&S has assisted Salem in support of restart efforts. In May 1995, Public Service Electric & Gas Co. (PSE&G) began drafting a starting lineup of DE&S system planners, system engineers and operations shift mentors. Teamed with existing PSE&G employees, DE&S is responsible for preparing Salem Units 1 and 2 for restart, obtaining all necessary regulatory approvals and assessing the plant's System Certification Process. DE&S has provided similar services to Carolina Power & Light's Brunswick Nuclear Station, Houston Power & Light's South Texas Project and New York Power Authority's Indian Point 3.

DE&S has developed state-of-the-art technologies and software to help streamline maintenance processes and meet the industry's continuous quest for safe, efficient and effective nuclear operations. DE&S provides full-scope preventive and predictive maintenance services, maintenance program optimization technologies and outage management services.

DE&S' Outage Maintenance Assessment (OMA) services help clients improve nuclear outage performance by reducing refueling outage duration and maintenance costs, while improving unit power availability. Using a computer software program, the OMA Analyzer, DE&S performs comprehensive analysis of nuclear outage maintenance work activities and tailors recommendations to meet plant-specific needs. DE&S has provided OMA services to Commonwealth Edison's LaSalle and Dresden nuclear stations, in addition to Duke Power plants.

DE&S is also helping clients design cost-effective preventive and predictive maintenance programs.

DE&S brings a high level of expertise in pumps, motors, breakers, wiring and valves to every predictive and preventive maintenance application.

DE&S has developed Maintenance Plusä, a comprehensive maintenance optimization process, which streamlines nuclear stations' processes and provides consistent implementation between nuclear sites. Supported by a software application, this program features a proven strategy for improving equipment reliability, maximizing process availability and reducing maintenance costs.

A.2 YAEC ASSET ACQUISITION

On December 1, 1997 DE&S completed the acquisition of the assets of YAEC's Nuclear Services Division (NSD). The NSD assets acquired consist mainly of the engineering and environmental services business previously conducted by YAEC. DE&S acquired substantially all of the assets of the NSD, including NSD employees. The remainder of YAEC is still a separate corporate entity with no change in ownership as a result of the acquisition. A brief description of the major departments formerly within YAEC NSD and their disposition is provided below.

NORTHEAST UTILITIES PROJECT - The NU Project was established to support Northeast Utilities demands. The project is staffed with electrical, mechanical, systems and I&C engineers mostly to perform modification design. A portion of these engineers have been permanently

transferred to Seabrook. The remaining engineers have been assigned to support various DE&S projects.

CONNECTICUT YANKEE PROJECT - The project consists of mechanical engineers, electrical engineers, I&C engineers, and systems engineers dedicated to the decommissioning of CY. The plans are to eventually move the staff to support the decommissioning of Connecticut Yankee

ENGINEERING SERVICES (ESD) - This is a multi-disciplined department set up by YAEC to sell engineering services to non-affiliate clients. Personnel within this department have been assigned to support other DE&S projects.

ENVIRONMENTAL ENGINEERING - This department has extensive experience in the areas of Environmental Sciences, Emergency Planning (EP), Radiological Engineering (RE) and Radiation Protection (RP) Consulting & Health Physics (HP) Site Services. The EP group provides centralized Emergency Planning services to Vermont Yankee, Maine Yankee, Seabrook and Pilgrim. The RE, RP and HP groups are also supporting decommissioning of Yankee Rowe and Connecticut Yankee. This department also provides Environmental Services for other New England non-nuclear utilities, other domestic and international nuclear utilities, the domestic commercial industries and the Federal Government.

ENVIRONMENTAL LABORATORY - This department works closely with Environmental Engineering and provides laboratory services in the following areas: (i) radiological sample analysis, (ii) in-vivo and in-vitro radio-bioassay, (iii) in-plant quality assurance program, (iv) personnel and environmental dosimetry monitoring, and (v) field and plant measurements

FUEL MANAGEMENT - The department provides nuclear fuel procurement, fuel management and economic analysis, spent fuel storage, fuel and component fabrication engineering, core performance assessment and special nuclear material accountability services.

MAINE YANKEE PROJECT - The staff assigned to the Maine Project have been permanently transferred to Maine Yankee.

NUCLEAR ENGINEERING - This is a specialized engineering department created to resolve technical issues in support of continued plant operation. The services provided include: core reload analysis, plant safety/performance analysis, cost/risk-benefit and reliability analysis and specialized thermal hydraulic analysis.

PLANT SUPPORT (PSD) - The services provided by PSD are: in-service inspections (ISI)/non-destructive examinations (NDE), fire protection, materials evaluation, welding, maintenance, system engineering, in-service testing, thermography, computer aided design (CAD) and construction.

VERMONT YANKEE PROJECT - All of the staff assigned to the Vermont Project have been transferred permanently to Vermont Yankee.

YANKEE ROWE PROJECT - This staff that supports the decommissioning of Yankee Rowe. The staff consists of systems engineers, licensing engineers, non-licensed operators, maintenance, security, HP, training personnel, etc.

The other departments acquired by DE&S that are not discussed above include: Information Resources, Treasury, Administrative/Security Services, and Health and Safety. All other aspects of YAEC, including ownership of Yankee Rowe, remains unaffected by the DE&S acquisition.

A.3 DE&S PROGRAM DESCRIPTION

This section summarizes the DE&S Quality Assurance Program, Design Control Program, and Project planning process

A.3.1 Quality Assurance Program

A.3.1.1 Organization

The DE&S organizational structure for activities affecting quality is shown in Figure A-2. The functional responsibilities, levels of authority, and lines of communication for the various organizational entities are described below.

The President, DE&S is responsible for quality and is the highest level of management responsible for establishing DE&S quality policies, goals, and objectives. The President, DE&S has documented DE&S' commitment to quality in the "Quality Assurance Program Policy Statement." The Quality Assurance Program Policy Statement is contained in the DE&S QA Program Description (QAPD) and the DE&S QA Procedures Manual.

The Business Unit Managers are responsible for technical quality achievement for nuclear work assigned to their business units under the controls of the DE&S QA Program. These Business Unit Managers report to the Executive Vice President who reports directly to the President and have full technical authority over activities affecting quality within their business unit.

The QA Manager reports to the President for quality assurance activities and is responsible for establishing and maintaining the DE&S Quality Assurance Program and verifying its effective implementation at all DE&S facility locations. The QA Manager is independent of managers responsible for performing quality affecting work and is sufficiently independent of cost and schedule considerations when in conflict with quality considerations. The QA Manager and President, DE&S jointly approve the DE&S QA Program Description and QA Procedures Manual. QA Program specific manuals (e.g., Standards Laboratory Operations Manual, Qualification and Testing Facility Reference Manual, Metallurgy Reference Manual, NDE Program Manual, Geotechnical Services Testing Manual, etc.) are jointly approved by the responsible Business Unit Manager and the QA Manager.

The QA Manager has direct access to the President and Business Unit Managers and is able to identify quality problems; initiate, recommend, or provide solutions; and verify implementation of solutions; and assure that further processing, delivery, installation, or use is controlled until proper disposition of a non-conformance, deficiency, or unsatisfactory condition has occurred.

The QA Manager has a quality assurance staff, consisting of auditors and support personnel. These individuals report directly to the QA Manager. QA personnel may be located at site offices; however, they remain in contact with the QA Manager for direction.

All DE&S personnel have the authority to stop work. The ultimate authority to stop or resume work resides jointly with the responsible Business Unit Managers and the QA Manager. Matters that cannot be resolved at this level are escalated to the President for resolution.

A.3.1.2 Program Description

The DE&S QA Program meets 10CFR50, Appendix B, 10CFR21, and ASME NQA-1-1989, through the NQA-1b-1991 Addenda. In addition, the DE&S QA Program can be tailored to meet other quality assurance requirements invoked by clients, such as ANSI N45.2, daughter standards of ANSI N45.2, and 10CFR72. The DE&S QA Program Description (QAPD) Manual describes DE&S commitments to 10CFR50 Appendix B, 10CFR21, and ASME NQA-1. The QAPD states policies, assigns responsibilities, and specifies requirements governing siting, design, construction, operation, and decommissioning of nuclear facilities. Specific processes and controls on these processes, which implement these commitments, are specified in QA Procedures contained in the QA Procedures Manual and in QA Program specific manuals that have been approved by the QA Manager.

The QA Program provides for the planning and accomplishment of activities affecting quality under suitably controlled conditions. Controlled conditions include the use of appropriate equipment, suitable environmental conditions for accomplishing the activity, and assurance that prerequisites for the given activity have been satisfied. The QA Program provides for any special controls, processes, test equipment, tools, and skills to attain the required quality and for verification of quality. To the extent necessary, requirements contained in this QAPD are invoked on DE&S suppliers and the supplier's subcontractors.

The DE&S QA Program is mandatory for all activities affecting quality. When work cannot be accomplished as specified in the implementing QA Procedure, or accomplishment of such work would result in an adverse condition, work is stopped until proper corrective action is taken.

A graded QA program is implemented for Quality Affecting Activities. Categories are defined as:

QA Condition 1 - Nuclear Safety Related

QA Condition 2 - Radioactive Waste Related

QA Condition 3 - Fire Protection Related (nuclear facilities)

QA Condition 4 - Seismic Category II Related

This QAPD describes controls for QA Condition 1. Alternate controls for QA Conditions 2, 3, and 4 are specified in implementing QA Procedures.

This quality assurance program may be used for non-nuclear projects as deemed necessary by the Business Unit Manager implementing the client contractual requirements.

A.3.1.3 Quality Assurance Training

Quality Assurance training is provided to all personnel performing activities affecting quality as determined by supervision. All DE&S permanent personnel, as well as loaned/part-time personnel who perform quality affecting activities, receive Quality Assurance Indoctrination Training. QA Indoctrination Training includes general criteria, including applicable codes, standards, QA Procedures, QA Program elements, and job responsibilities and authorities. Detailed QA training is provided on the QA Program, applicable Project Plans and procedures prior to an employee beginning work affecting quality. Supervision is responsible for assuring personnel performing work under their supervision are trained.

The QA Training Coordinator (QATC) is appointed by the President, DE&S and is responsible for coordinating QA training activities for DE&S. The QATC serves as a centralized training support service for supervision in coordinating training and maintaining QA training records. This responsibility is carried out as support for line management. Appropriate DE&S supervisory personnel are responsible for determining the type and extent of the training to be provided to an individual, and ensuring that the training is properly documented for personnel performing activities affecting quality.

A.3.1.4 Management Assessments

The President conducts a Corporate Management Assessment every two years to determine if the Quality Assurance Program is effective on a corporate-wide basis. The President appoints a team of DE&S managers or supervisors to conduct this assessment. Recommendations are provided to the President for action.

Business Unit Managers and the QA Manager conduct an Internal Management Assessment annually of QA activities under their control. The managers report results of the Internal Management Assessments to the President for review.

A.3.1.5 Qualification/Certification of Nondestructive Examination (NDE) Personnel

DE&S maintains an NDE program described in the NDE Program Manual and may also use an approved outside agency to provide training and to administer examinations for certification by DE&S of qualified NDE personnel. In general, DE&S uses Duke as the approved outside agency. The ASME Recommended Practice No. SNT-TC-1A, Personnel Qualification and Certification in Nondestructive Testing, 1984 Edition applies as requirements for NDE personnel.

A.3.1.6 QA Program Reporting to Management

Management is regularly informed on the quality of DE&S quality affecting work via results of audit reports, internal surveillance reports, Problem Investigation Reports, and management assessments. Corrective action is initiated and monitored until completion as necessary.

A.3.1.7 Qualification/Certification of Inspection and Test Personnel

DE&S uses an approved outside agency to provide training and to administer examinations for certification by DE&S of qualified inspection and test personnel. In general, DE&S uses Duke as the approved outside agency. Inspection and test personnel are qualified and certified in accordance with ANSI/ASME N45.2.6 - 1978, Qualifications of Inspection, Examination, and Test Personnel.

A.3.1.8 Qualification/Certification of Personnel for ASME Code Design

Qualifications and certification of personnel (Registered Professional Engineers) engaged in certifying ASME Section III Code designs and documents, when required by ASME Section III, is conducted in accordance with ANSI/ASME N626.3, Qualifications and Duties of Specialized Professional Engineers.

A.3.2 Design Control Program

Measures are established in QA Procedures to assure that applicable requirements are correctly translated into design documents. Design inputs are specified. Controls are established for the selection and suitability of application of materials, parts, equipment, and processes that are essential to the functions of structures, systems, and components. Controls are established for the identification of design interfaces among participating organizations. QA Procedures specify controls for design input, design process, design verification, design changes, and approval. These procedures include appropriate quantitative and/or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Design documents are prepared, verified, and approved by qualified individuals. Design verification is performed by individuals who are independent of the preparer or approver. Design is verified by one or more of the following verification methods: design reviews, alternate calculations, or qualification tests. The method of design verification and results are documented. Design changes are governed by control measures commensurate with those applied to the original design. Computer

software is verified and validated in accordance with the requirements of ASME NQA-2, Part 2.7 and ASME NQA-1, Supplement 11S-2.

A.3.2.1 Design Process Description

Applicable design inputs (such as design bases, conceptual design reports, performance requirements, regulatory requirements, codes, and standards) are controlled in accordance with the following requirements:

- Design inputs are identified and documented, and their selection reviewed and approved by responsible management.

- Design inputs are specified and approved to the level of detail necessary to permit the design work to be carried out in a correct manner that provides a consistent basis for making design decisions, accomplishing design verification, and evaluating design changes.

- Changes from approved design inputs and reasons for the changes are identified, approved, documented, and controlled. Design inputs based on assumptions that require reverification are identified and controlled.

The design process is controlled. Design work is prescribed and documented on a timely basis and to the level of detail necessary to permit the design process to be carried out in a correct manner. Design documents are adequate to support design, fabrication, construction, test, inspection, examination, and operation. Appropriate standards are identified and documented, and their selection reviewed and approved. Changes from specified standards, including the reasons for the change, are identified, approved, documented, and controlled. Controls for selecting and reviewing design methods, materials, parts, equipment, and processes that are essential to the function of an item for suitability of application are established. Applicable information derived from experience, as set forth in reports or other documentation, is made available to cognizant design personnel. Design documents are sufficiently detailed as to purpose, method, assumptions, design input, references, and units such that a person technically qualified in the subject can understand the documents and verify their adequacy without recourse to the originator. Controls for identifying assemblies or components that are part of the item being designed are established. When such an assembly or component is a commercial grade item, that prior to its installation, is modified or selected by special inspection and/or testing to its requirements that are more restrictive than the supplier's published product description, the component part is represented as different from the commercial grade item in a manner traceable to a documented definition of the difference. Drawings, specifications, or other design output documents shall contain appropriate inspection, examination, and testing acceptance criteria.

Design analyses are planned, controlled, and documented. Design analysis documents are legible and in a form suitable for reproduction, filing, and retrieval.

Calculations are identifiable by subject (including structure, system, or component to which the calculation applies), originator, reviewer, and date, or by other designators such that the calculations are traceable. Computer software used to perform design analyses is developed or qualified, and used according to the requirements of ASME NQA-2, Part 2.7 and ASME NQA-1 Supplement 11S-2. Documentation of design analyses includes definition of the objective of the analyses, definition of design inputs and their sources, results of literature searches or other applicable background data, identification of assumptions and designation of those that must be verified as the design proceeds, identification of any computer calculation, including computer type, computer program (e.g., name), revision identification, inputs, outputs, and the bases (or reference thereto) supporting application of the computer program to the specific physical problem, review and approval, identification of analysis methods utilized, identification of the design/analysis results and demonstration that they meet the applicable acceptance criteria, and the conclusion of the design/analysis.

A.3.2.2 Design Verification

Design verification is performed using one or a combination of the following methods: design review; alternate calculations; or qualification testing. The particular design verification method is documented. The results of design verification are documented, including the identification of the verifier. Design verification is performed by competent individuals or groups other than those who performed the original design but may be from the same organization.

Design verification is performed at appropriate times during the design process. Verification is performed before release for procurement, manufacture, or construction or release to another organization for use in other design work. In some cases (such as when insufficient data exists) it may be necessary to release unverified designs to other organizations to support schedule requirements. Unverified portions of the design are clearly identified and controlled. In all cases, design verification is completed before relying on the item to perform its function. The extent of the design verification required is a function of the importance to safety, complexity of design, degree of standardization, state of the art, and similarity with previously proven designs. Use of previously proven designs is controlled. The applicability of standardized or previously proven designs are verified with respect to meeting pertinent design inputs for each application. Known problems affecting standard or previously proven designs and their effects on other features is considered. The original design and associated verification measures are adequately documented and referenced in the files of subsequent application of the design. Changes in previously verified designs shall require re-verification. Such verifications shall include the evaluation of the effects of those changes on the overall previously verified design and on any design analyses upon which the design is based.

Design reviews are controlled and are focused on assuring that: (i) design inputs were correctly selected and incorporated, (ii) assumptions necessary to perform

the design work were adequately described, reasonable, and, where necessary, reverified (iii) appropriate design methods were used , (iv) design outputs are reasonable compared to design inputs, and (v) necessary design inputs and verification requirements for interfacing organizations were specified in the design documents or in supporting implementing documents.

Alternate calculations focus on evaluating the appropriateness of assumptions, input data, and the computer program or other calculation method used, and the results verify the correctness of the original calculations or analyses.

If design adequacy is to be verified by qualification testing, the tests are identified, controlled, and documented. The test configuration is defined and documented. Testing shall demonstrate the adequacy of performance under conditions that simulate the most adverse design conditions. Operating modes and environmental conditions in which the item must perform satisfactorily are considered in determining the most adverse design conditions. If the tests verify only specific design features, then the other features of the design are verified by other means. Test results are documented and evaluated to ensure that test requirements have been met. If qualification testing indicates that a modification to an item is necessary to obtain acceptable performance, then the modification is documented and the item modified and retested or otherwise verified to ensure satisfactory performance. Scaling laws are established, verified, and documented when tests are being performed on models or mockups. The results of model test work are subject to error analysis, where applicable, before using the results in final design work.

A.3.2.3 Design Changes

Design changes are controlled. Changes to final designs, field changes, and nonconforming items dispositioned "use-as-is" or "repair" are justified and are subject to design control measures commensurate with those applied to the original design. Design control measures for changes shall include provisions to ensure that the design analyses for the item are still valid. Changes are approved by the same affected groups or organizations that reviewed and approved the original design documents, unless otherwise approved by DE&S management. The design process and design verification methods and implementing documents are reviewed and modified, as necessary, when a significant design change is necessary because of an incorrect design. These design deficiencies are documented and resolved in accordance with approved procedures.

Field changes are incorporated into affected design documents when such incorporation is appropriate, and when a field change is approved other than by revision to the affected design documents. Design changes that impact related implementing documents or training programs are communicated to affected organizations.

Design interfaces are identified and controlled. Design efforts are coordinated among interfacing organizations. Interface controls shall include the assignment of responsibility and the establishment of implementing documents among interfacing design organizations for the review, approval, release, distribution, and revision of documents involving design interfaces. Design information transmitted across interfaces are documented and controlled. The status of the design information or document provided is identified in transmittals. Where necessary, incomplete designs that require further evaluation, review, or approval are identified. When it is necessary to initially transmit the design information orally or by other informal means, design information is promptly confirmed through a controlled implementing document

The QA Manager shall review design documents to assure inclusion of the applicable quality requirements as specified in implementing quality assurance procedures.

A.3.2.4 Software Utilization

Computer Software is verified and validated. Software verification and validation activities shall ensure that the software adequately and correctly performs all intended functions; and ensure that the software does not perform any unintended function that either by itself or in combination with other functions can degrade the entire system. Software verification and validation activities are planned and performed for each system configuration that may impact the software. The results of software verification and validation activities are documented. Software verification and validation are performed by individuals other than those who designed the software. Software verification is performed during the software development to ensure that the products of a given cycle phase fulfill the requirements of the previous phase or phases. Applicable software life cycle phases are as defined by ASME NQA-2 Part 2.7.

Software validation is performed at the end of the implementation phase to ensure that the code satisfies the requirements. Software validation activities, such as the development of test plans and test cases are integrated into each phase of the software life cycle. Testing is the primary method of software validation. Software testing is conducted in accordance with ASME NQA-1 Supplement 11S-2. The validation of modifications is subject to selective regression testing to detect errors introduced during the modification of systems or system components, to verify that the modifications have not caused unintended adverse effects, or to verify that a modified system(s) or system component(s) still meets specified requirements.

A configuration baseline is defined at the completion of each major phase of the software development. Approved changes created subsequent to a baseline are added to the baseline. A baseline shall define the most recent approved software configuration.

A labeling system for configuration items is implemented. The labeling system covers: uniquely identifies each configuration item; identifies changes to configuration items by revision; and provides the ability to uniquely identify each configuration of the released software available for use.

Changes to software are formally documented. This documentation shall contain a description of the change, the rationale for the change, and the identification of affected baselines. The change is formally evaluated and approved by the organization responsible for the original design, unless an alternate organization has been given the authority to approve the changes. Only authorized changes are made to software baselines. Software verification activities are performed for the change as necessary to ensure the change is appropriately reflected in software documentation, and to ensure that document traceability is maintenance. Software validation is performed as necessary for the change.

The information that is needed to manage a configuration is documented. This information shall identify the approved configuration, the status of proposed changes to the configuration, the Status of approved changes, and information to support the functions of configuration identification, and configuration control.

A plan(s) for assuring software quality assurance is in existence for each new software project at the start of the software life cycle, or for procured software when it enters the purchaser's organization. This plan(s) may be prepared individually for each software project, or may exist as a generic document to be applied to software prepared within or procured by an organization, or may be incorporated into the overall quality assurance program. The plan for software quality assurance shall identify: the software products to which it applies; the organizations responsible for performing the work and achieving software quality and their tasks and responsibilities; required documentation; standards, conventions, techniques, or methodologies which shall guide the software development, as well as methods to assure compliance to the same; the required software reviews; and the methods for error reporting and corrective action.

Software requirements documentation outline the requirements that proposed software must satisfy. These requirements address, as applicable, the following: (i) functionality - the functions the software is to perform, (ii) performance - the time-related issues of software operation such as speed, recovery time, and response time, (iii) design constraints imposed on implementation phase activities - any elements that will restrict design options, (iv) attributes - non-time-related issues of software operation such as portability, acceptance criteria, access control, and maintainability, and (v) external interfaces - interactions with people, hardware, and other software. An item can be defined as a software requirement only if its achievement can be verified and validated. Software requirements are traceable throughout the remaining stages of the software development cycle.

Software design and implementation documentation includes a document or series of documents that shall contain: a description of the major components of the

software design as they relate to the software requirements; a technical description of the software with respect to the theoretical basis, mathematical model, control flow, data flow, control logic, and data structure; a description of the allowable or prescribed ranges for inputs and outputs; and the design described in a manner that can be translated into code.

A.3.2.5 Software Verification

Software verification and validation documentation shall describe the tasks, and criteria for accomplishing the verification of the software in each phase, and the validation of the software at the end of the development cycle. The documentation shall also specify the hardware and software configurations pertinent to the software verification and validation. The documentation is organized in a manner that allows traceability to both the software requirements and the software design. This documentation shall also contain the results of the execution of the software verification and validation activities, and shall include the results of reviews and tests, and a summary of the status of the software e.g., incomplete design performance and application requirements.

User documentation, as a minimum, shall include: user instructions that contain an introduction, a description of the user's interaction with the software, and a description of any required training necessary to use the software; input and output specifications; input and output formats; a description of system limitations; a description of anticipated errors and how the user can respond; and information for obtaining user and maintenance support.

Verification reviews shall identify the participants and their specific responsibilities during the review and in the preparation and distribution of the review documentation. The reviewed documents are updated and placed under configuration control. Documentation of review comments and their disposition is retained until they are incorporated into the updated software. Comments and their disposition not incorporated are retained in accordance with established procedures. The review of software requirements is performed at the completion of the software requirements documentation. This review shall assure that the requirements are complete, verifiable, consistent and technically feasible. The review shall also assure that the requirements will result in feasible and usable code. The software design review is held at the completion of the software design documentation. This review shall meet the design verification requirements of ASME NQA-1, 3S-1. This review shall evaluate the technical adequacy of the design approach, and assure internal completeness, consistency, clarity, and correctness of the software design, and shall verify that the software design is traceable to the requirements. Upon completion of the testing phase (and the installation phase if necessary) the development cycle documentation is reviewed and approved to assure completion and acceptability.

A formal procedure of software problem and corrective action is established for software errors, and failures. This problem reporting system shall assure that

problems are promptly reported to affected organizations to assure formal processing of problem resolutions. Problems in software may be classified by the organization responsible for the evaluation. Any classification system shall have defined criteria based on the impact of the software output. Corrective action by the responsible organization shall assure that: problems are identified, evaluated, documented, and, if required corrected; problems are assessed for impact on past and present applications of the software by the responsible organization; corrections or changes are controlled.; and preventive actions and corrective actions results are provided to affected organizations.

A.3.2.6 Software Control

Controls are established to permit authorized and prevent unauthorized access to a computer system. Individuals or organizations developing and supplying software under contract are required to have policies and procedures that meet the applicable requirements of this Part as specified in procurement documents. The documentation that is required by this Part is delivered or made available by the supplier to the purchaser. The applicable requirements of this section are the responsibility of the purchaser upon receipt of software. Typically this software enters the purchaser's organization at the start of the installation and checkout phase. The supplier shall report software errors, or failures, to the purchaser, and the purchaser shall report software errors to the supplier.

Software that has been developed not using ASME NQA-2, Part 2.7 is placed under the configuration controls required by ASME NQA-2, Part 2.7 prior to use. The user organization shall perform and document an evaluation to: determine its adequacy to support software operation and maintenance, and identify the activities to be performed and documents that are needed in order for the software to be placed under configuration control. This determination is documented and shall identify as a minimum: user application requirements; test plans and test cases required to validate the software for acceptability; user documentation. After the specified activities are performed, reviewed, and approved, the software is placed under configuration control.

As an alternate, the user organization shall obtain the above documentation from the supplier or perform a documented review of the documentation at the supplier facility to determine acceptability. This review shall meet the requirements as specified above.

The organization providing' software services shall have a plan(s) for software quality assurance that meets the requirements of Section 10. The user organization shall determine the adequacy of this plan.

A.3.2.7 Software Testing

Computer Program Test requirements and acceptance criteria are provided or approved by the organization responsible for the design or use of the program to

be tested unless otherwise designated. Required tests including (as appropriate) verification tests, hardware integration tests, and in-use tests are controlled. Test requirements and acceptance criteria are based upon applicable design or other pertinent technical documents. Verification tests shall demonstrate the capability of the computer program to produce valid results for test problems encompassing the range of permitted usage defined by the program documentation. Acceptable test problem solutions are as follows: hand calculations; data and information from technical literature. For programs used for operational control, testing shall demonstrate required performance over the range of operation of the controlled function or process. Depending on the complexity of the computer program being tested, testing may range from a single test of the completed computer program to a series of tests performed at various stages of computer program development to verify correct translation between stages and proper working of individual modules, followed by an overall computer program test. Regardless of the number of stages of testing performed, verification testing is sufficient to establish that test requirements are satisfied and that the computer program produces a valid result for its intended function. Test problems are developed and documented to permit confirmation of acceptable performance of the computer program in the operating system. Test problems are run whenever the computer program is installed on a different computer, or when significant hardware or operating system configuration changes are made. Periodic in-use manual or automatic self-check routines are prescribed and performed for those applications where computer failures or drift can affect required performance.

Test procedures or plans shall specify the following, as applicable: required tests and test sequence; required ranges of input parameters; identification of the stages at which testing is required; criteria for establishing test cases; requirements for testing logic branches; requirements for hardware integration; anticipated output values; acceptance criteria; and reports, records, standard formatting, and conventions.

Test results are documented. Verification test results are evaluated by a responsible authority to assure that test requirements have been satisfied.

Verification test records shall identify: computer program tested; computer hardware used; test equipment and calibrations, where applicable; date of test; tester or data recorder; simulation models used, where applicable; test problems; results and acceptability; action taken in connection with any deviations noted; and person evaluating test results

In-use test results shall identify: computer program tested; computer hardware used; test equipment and calibrations, where applicable; date of test; tester or data recorder; and acceptability

A.3.3 Project Planning Process

Project planning is performed to specify the methods, controls, and procedures by which a scope of work, and the client specified technical and quality assurance requirements, are to be accomplished. Each client project performed under the DE&S QA Program requires preparation of a Project Plan. The Project Plan defines the applicable QA procedures and requirements, including the identification of specific QA Program manuals, client procedures, or other client specified project requirements to the project. Specific time limits are placed on the preparation and approval of a Project Plan with respect to authorization of work performance. The level of detail of the project plan is commensurate with the scope and complexity of the project. The Project Plan specifies, as a minimum:

- QA Condition (e.g., QA Condition 1, QA Condition 2, QA Condition 3, or QA Condition 4);
- 10CFR21 Applicability (Do 10CFR21 requirements apply to the project?);
- Project Description;
- QA Procedure identification (Identify specific QA procedures that apply to the project.);
- Interface Identification (identify internal/external project interfaces and controls utilized to ensure that the proper project interface is achieved.);
- Software listing (Identify software and version to be used on the project);
- Deliverable listing (List deliverables, such as specifications and calculations, to be prepared as part of the project.);
- Design Input Definition (Identify source documents or list applicable design characteristics and/or functions to be used in developing design inputs, including applicable codes, standards and regulatory requirements.);
- Experience Information (Identify previous experience which is pertinent to the project, including lessons learned from similar projects);
- Scope of work (Provide detailed description of work to be performed);
- Special requirements (Identify special personnel certification requirements, client specified Codes / Standards, restrictions for Suppliers, special shipping/handling/storage requirements, etc.);

Documented training is required for all project personnel on the requirements and procedures specified within the Project Plan. Preparation, control, revision, and close-out of Project Plans are governed by DE&S procedure. DE&S Project Plans are defined as QA records.

FIGURE A-1

Duke Energy Corporation Organization

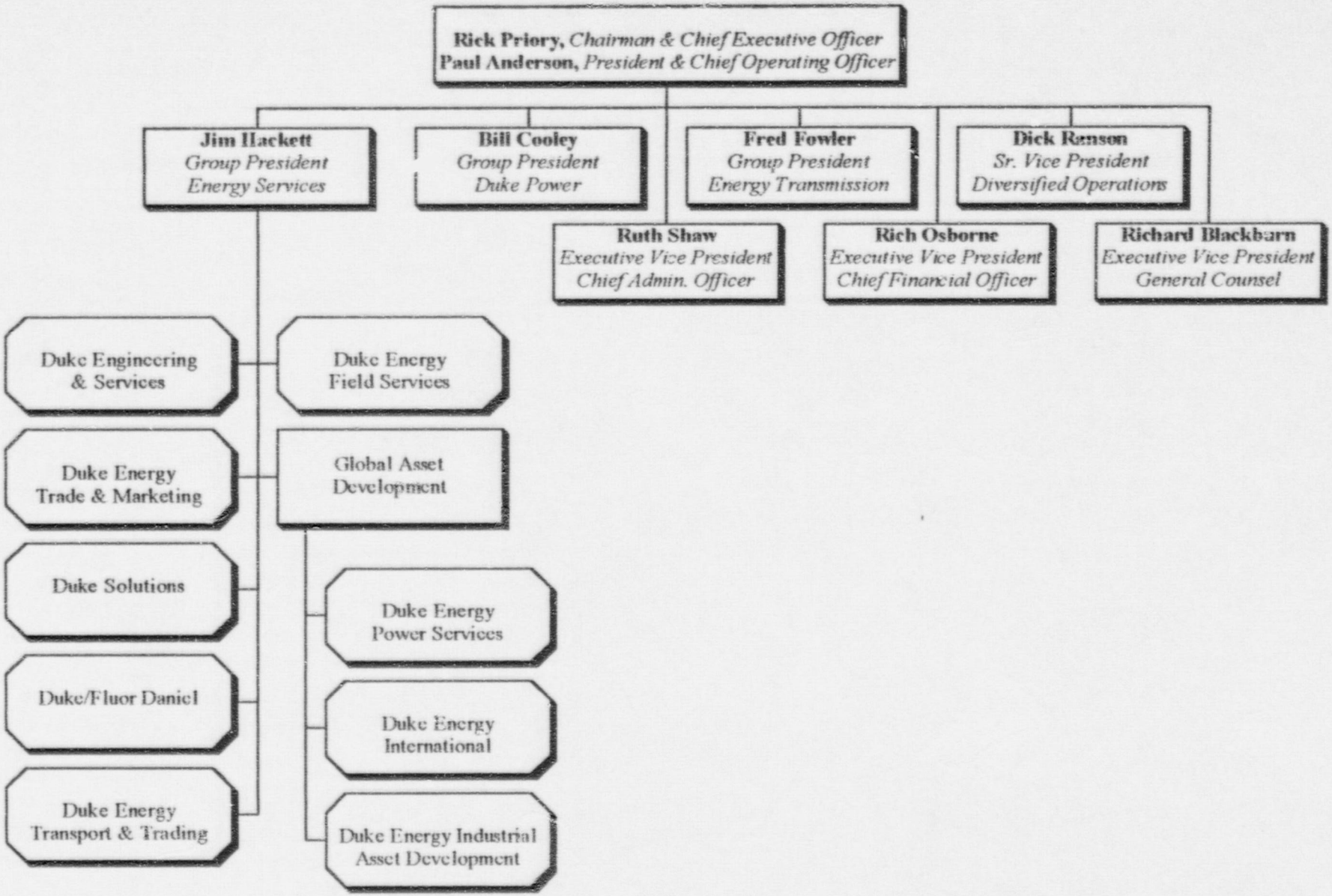
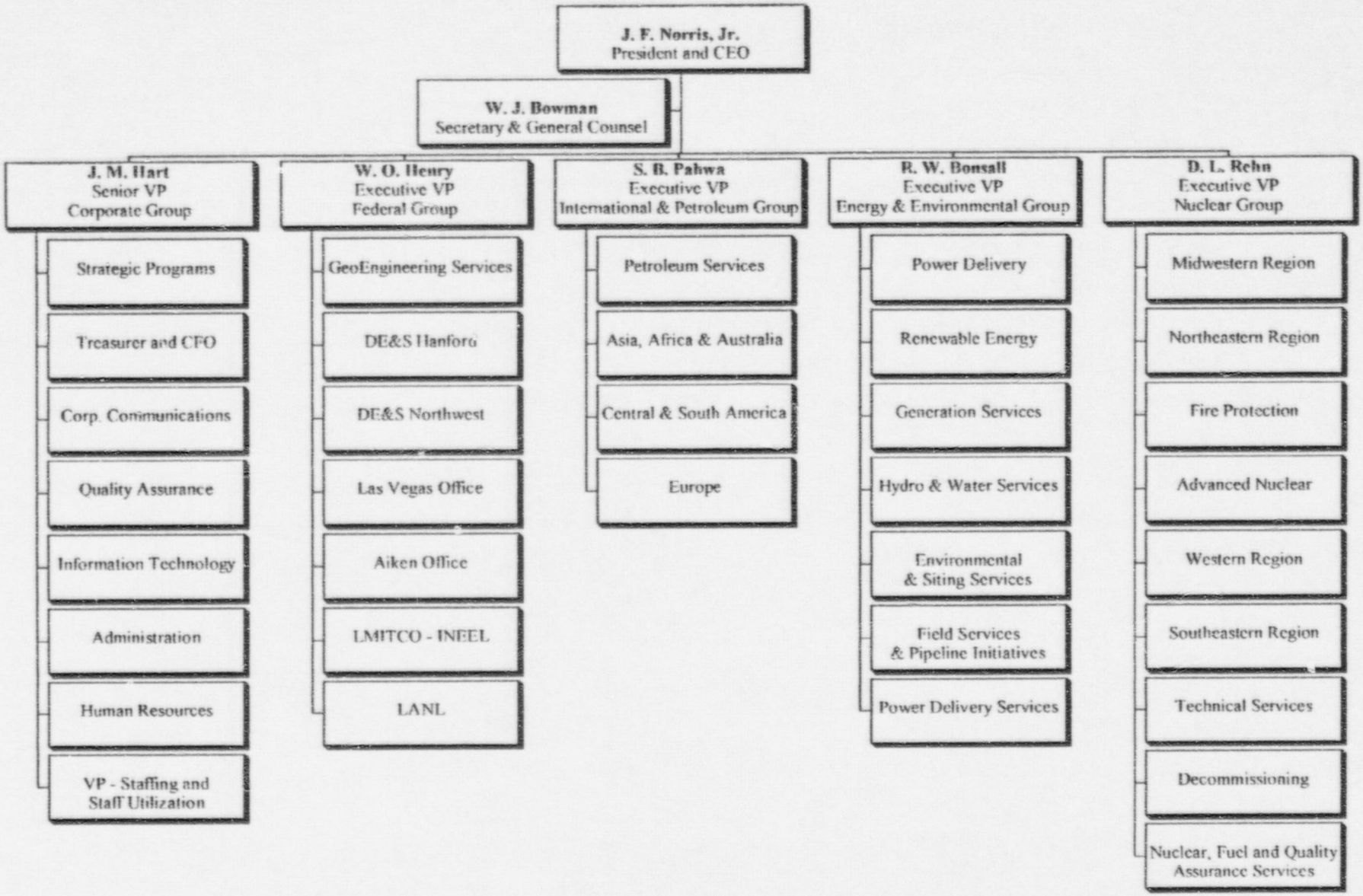


FIGURE A-2

Duke Engineering & Services, Inc. Organization



APPENDIX B

DESCRIPTION OF YAEC RESPONSE TO SAFETY ALLEGATIONS

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APPENDIX B

DESCRIPTION OF YAEC RESPONSE TO SAFETY ALLEGATIONS

Upon notification of the safety allegations transmitted by UCS in December 1995, YAEC promptly initiated a technical review of the allegations. Further YAEC review and interaction with the NRC, including the NRC Independent Safety Assessment for Maine Yankee and the RELAP5YA (BWR) LOCA applications audit performed in the summer of 1996, led to a number of technical/programmatic assessments and corrective actions. This appendix summarizes the more significant of these assessments and actions taken by YAEC prior to the December 1997 DE&S acquisition. Details of actions summarized in this appendix are generally found in Maine Yankee Atomic Power Station NRC licensing docket, Docket 50-309.

This appendix is not to be construed as a response by YAEC to the Demand. This appendix only represents a DE&S summarization of actions taken by YAEC. This summary of YAEC actions is provided to serve as an aid to the reader in understanding the actions taken by and conclusions of DE&S in responding to the Demand.

B.1 TECHNICAL ASSESSMENT

B.1.1 January 1996 YAEC Assessment

The Maine Yankee safety allegations transmitted in December 1995 immediately prompted MYAPCo and YAEC executive management to form two corporate teams to evaluate the allegations: (1) a Response Team composed of managers and technical specialists with responsibilities for the analyses being questioned and (2) an Independent Review Team compose of managers and technical specialists with no prior responsibilities for these analyses. The Response Team mission was designed to provide support for interaction with the NRC in reviews, as well as to evaluate the allegations, identify and resolve technical issues, and to ascertain whether any issues raised by the allegations or in subsequent reviews adversely affected the conclusions of the safety analyses used to support past or future operation of the Maine Yankee plant. The Independent Review Team was also charged with assessing the impact of the allegations on the current licensing basis and the safety significance of the issues.

The Response Team supported the NRC inspection performed in response to the allegations during the week of December 11-14, 1995 and provided additional information to the NRC technical staff. This team, along with executive management and other representatives from MYAPCo and YAEC, met with NRR management and technical staff on December 18, 1995. The proceedings of these interactions with the NRC Staff demonstrated clearly that a lengthy breakdown in effective communications between YAEC and the Staff had occurred. The review status of the Maine Yankee SBLOCA application did not meet NRC expectations. In retrospect, Yankee should have increased its direct communications with the NRC following the approval of RELAP5YA(PWR) for Maine Yankee application in 1989.

Within the next four weeks the YAEC technical assessment of the suitability of using RELAP5YA(PWR) for demonstrating conformance with the criteria set forth in 10CFR50.46 was completed and provided to the NRC in Reference B.1. In that submittal YAEC concluded that the results obtained using RELAP5YA(PWR) were conservative and that SER conditions had been met.

In the January 1996 submittal, YAEC stated that the Maine Yankee ECCS performance during postulated SBLOCA conditions was acceptable and met all NRC regulations. YAEC also stated that during development of computer codes to analyze these conditions, some results were generated that, if valid, would not meet NRC requirements. However, these results were indicative of problems with the modeling used in development of the computer code and the Maine Yankee plant model. When a more appropriate plant model was developed, YAEC believed that the results demonstrated that the Maine Yankee ECCS performance was adequate and all NRC requirements were met. Thus YAEC concluded that the operating limits for the plant were properly set on the basis of the limiting LBLOCA analysis using NRC approved methods. LBLOCA was consistently the historical LOCA licensing basis for Maine Yankee. Subsequent analyses that were performed by Siemens (Section B.1.2) reconfirmed this conclusion that the safety limits are conservative when established on the basis of the LBLOCA analysis.

The computer code RELAP5YA(PWR)[B.2] was approved by the NRC for use as a licensing method in evaluating the performance of the Maine Yankee ECCS under SBLOCA conditions.[B.3] RELAP5YA(PWR) is a modified version of RELAP5/MOD1,[B.4] a commonly used computer code that was developed by the Idaho National Engineering Laboratory over the course of several years through the early 1980's. The RELAP5YA(PWR) code was submitted for NRC review in 1983.[B.5] NRC employed the technical assistance of the Los Alamos National Laboratory[B.6] and the Idaho National Engineering Laboratory[B.7] in the review of the topical report. NRC requested responses to 197 questions on general application of the code in May 1984 and asked several additional questions on the SBLOCA application in September 1986. A series of eight submittals responding to these questions was transmitted to the NRC between March 1984 and December 1986. A final submittal on SBLOCA application model improvements with appropriate comparisons to experimental test results was made in 1988. These submittals were supported by numerous meetings and discussions with the staff and their consultants and culminated in approval of the code for application to Maine Yankee in January 1989. In addition, the NRC staff and consultants had completed in 1986 the review and approval of RELAP5YA(PWR) analysis which was performed to provide guidance on Reactor Coolant Pump operation during SBLOCA transients for Maine Yankee.[B.8,B.9]

The NRC approval of RELAP5YA(PWR) in 1989 included twelve specific conditions to be met in the use of this code for Maine Yankee. Several of these conditions specified that the code was required to be used in a certain manner. For instance, one of the conditions requires the code to be used assuming an emergency core cooling system water temperature of 200 °F. These conditions were met in the Maine Yankee application.[B.10,B.11] Five conditions relate to information to be provided in plant specific licensing applications that the NRC would review to examine how the code was used. The Response Team found that these submittals had not been made based on the understanding of a later NRC letter which

indicated that this information should be retained for inspection.[B.12] The Response Team verified the information was available for inspection by the NRC.

During the inspection the week of December 11, 1995, the NRC raised a number of questions about the application of RELAP5YA(PWR) to the Maine Yankee SBLOCA analysis which could not be resolved satisfactorily during the four day review. These questions related to the fluctuations (oscillations) in peak clad temperature (PCT) calculated by the code, the time step sizes used in the analysis, the flow loss coefficient (K-factor) developed and used in the analysis, and the range of break sizes analyzed. References B.1 and B.13 provided information addressing these questions with the intent of demonstrating that the RELAP5YA(PWR) application to Maine Yankee provided a conservative calculation for the PCT values for SBLOCA analysis.

Reference B.1 also summarized additional analyses that had been performed using an advanced version of RELAP5-MOD 3 that further supported the conclusions that had been reached in the RELAP5YA(PWR) analysis of Maine Yankee. However, efforts to justify the suitability of RELAP5YA use for Maine Yankee SBLOCA analyses were very limited subsequent to the submittal of Reference B.1. MYAPCo decided to retain Siemens to conduct an independent analysis of SBLOCA for Maine Yankee.

B.1.2 Siemens Analysis of Maine Yankee SBLOCA

In Reference B.14 MYAPCo submitted analyses for SBLOCA analyses conducted by Siemens. The Siemens SBLOCA analyses were performed to support a nominal power of 2700 MWt. Features included a conservative local power assumption and variations in axial power profiles. RCP trip sensitivity studies were also performed. The analyses were performed to demonstrate that ECCS acceptance criteria, as stated in 10CFR50.46, were met.

Break spectrum calculations were performed to identify the limiting break size. Break sizes of 0.05, 0.10, 0.15, 0.20, 0.25, and 0.61 ft^2 in the pump discharge side of a cold leg pipe were analyzed. An axial profile peaked at a relative core height of 73% was used in the break spectrum analysis. No core heatup was calculated for the 0.05 ft^2 break size. The 0.10 ft^2 break size resulted in the highest Peak Cladding Temperature (PCT) for the break spectrum cases.

Additional axial profile calculations were performed at the limiting break size for axial profiles peaked at relative core heights of 52%, 65%, and 85%. The 65% axial profile case gave the highest PCT for the axial profile sensitivities. Since the PCTs for the axial profiles peaked at relative core heights of 65%, 73%, and 85% were relatively close, core cross flow loss coefficient sensitivity calculations, as required by the Siemens SBLOCA methodology, were performed for these three axial profile cases. The limiting PCT was for the axial profile peaked at a relative core height of 73% with the minimum core cross flow loss coefficient. The PCT was 1781°F.

All 10CFR50.46 criteria were demonstrated to be met for the limiting case:

1. The limiting PCT was calculated to be 1781°F, and was for IFBA fuel. The margin to the 10CFR50.46 PCT limit was shown to be in excess of 400°F.

2. The local cladding oxidation was calculation to be 1.5 percent.

3. The core-wide metal-water reaction was calculated to be less than 1 percent.

4. The core remains amenable to cooling by staying within the local oxidation criteria.

5. Flow rates from the Emergency Core Cooling System (ECCS) ensure that the core temperatures have been reduced to acceptably low values and long-term cooling has been established.

The Reactor Coolant Pump (RCP) trip delay time evaluation consisted of both Evaluation Model (EM) analyses and best estimate analyses. The EM evaluation bounded RCP trip delay times of up to two minutes and demonstrated that 10CFR50.46 criteria are satisfied. The limiting PCT for this analysis was calculated to be 1781°F, and was for IFBA fuel. Best estimate calculations were performed to support RCP trip delay times from two minutes to 10 minutes. Break spectrum calculations were performed for break sizes of 0.10, 0.15, 0.25, 0.45, and 0.61 ft^2. Several RCP trip delay times were chosen for each break size calculation to assure that a bounding case was analyzed for each break size. The limiting best estimate case was determined to be the 0.15 ft^2 break, 10-minute RCP trip delay time case, with a PCT of 1136°F.

The NRC had not completed its review of the Siemens analysis when MYAPCo decided to end plant operation. YAEC was unaware of any outstanding issues, however, that would have changed the overall conclusion reached by Siemens in this analysis.

The results from these Siemens analyses were similar to those determined with RELAP5YA with regard to important analysis trends and conclusions. The PCT calculated by Siemens was lower than determined with RELAP5YA, and thus supported YAEC conclusions that the RELAP5YA (PWR) results were conservative and well bounded by LBLOCA analyses.

B.1.3 NRC Independent Safety Assessment

B.1.3.1 Assessment Description and Summary of Conclusions

The NRC Independent Safety (ISA) review for Maine Yankee was conducted in response to findings developed by the NRC Office of the Inspector General (OIG) in a report dated May 8, 1996. The ISA review team and approach was developed at the direction of the Chairman of the NRC to provide a thorough and substantial examination of all safety-related activities which supported Maine Yankee operation. One group from the 25-member ISA team was assigned for the duration of the effort to perform an assessment of the analytic code support provided for Maine Yankee YAEC. The approach and results of this review of YAEC safety-related analysis activities are described in this section.

The ISA team was led by a senior NRC manager, and included representatives of the State of Maine. To ensure an independent perspective, the NRC members were selected from NRC offices other than the Office of Nuclear Reactor Regulation (NRR) and NRC Region I offices. The team management reported the results of this assessment directly to the Chairman of the NRC.

The ISA team conducted their reviews from June through October in 1996. The portion of the team that reviewed YAEC activities in the Bolton office included four full-time NRC staff and expert industry consultants who performed on-site review activities for four weeks over a calendar period of six weeks. In addition the team performed documentation reviews, benchmarking analysis, and assessment activities in NRC White Flint offices both prior and subsequent to these face-to-face interactions. The scope of the examination included essentially all safety-related analyses, methods, computer codes, and procedures associated with work performed to support Maine Yankee activities, including that which was performed by the YAEC LOCA Group. Due to the on-going NRC Office of Investigation (OI) review, the only work that was exempt from the review was that which had been used specifically to develop the LOCA analysis limits for Maine Yankee.

The final report of the full NRC ISA team was transmitted to Maine Yankee Atomic Power Company on October 7, 1996. The comments that follow were derived directly from an integration of all comments identified in the report that relate to the review of the Nuclear Engineering Department safety-related activities. As was stated in the ISA final report, this particular area of assessment (that is, review of the use and application of analytic codes) had not been typically reviewed as part of the NRC regulatory process. Consequently, a panel of acknowledged experts in the area of code development and phenomenology was assembled by the NRC to provide a critical review the findings and observations of the ISA team in this area. Findings regarding this aspect of the ISA team review as described in the ISA report were summarized as follows.

The use of analytic codes for safety analyses was found to be very good. Cycle specific core performance analyses were found to be excellent. More complicated, less frequently performed safety analyses contained weaknesses, but the analyses were found to be acceptable based on compensating margin. YAEC did not have a written process to document how safety analyses conformed to SER conditions. Some conditions were clearly known, considered, and used by YAEC. Other conditions could not be shown to be satisfied until additional analyses, assessments, and sensitivity studies were accomplished in response to ISA team requests. This new work demonstrated that all SER conditions had been satisfied, although the disposition of some issues required reliance on the known conservatism in specific accident analyses.

B.1.3.2 Approach to Assessment

The team evaluated the analytic code support provided by YAEC for MYAPCo to assure that Maine Yankee was operated within the bounds of the safety analyses. This assessment was performed by reviewing the YAEC process for conducting non-LOCA safety analyses described in Chapter 14 of the Maine Yankee FSAR, and by performing an in-depth review of two specific safety analyses: the Control Element Assembly (CEA) drop transient and the steam line rupture accident. Selection of the dropped CEA transient for in-depth review provided a structured means to examine many of the computer codes used by YAEC. Selection of the steam line rupture analysis provided a forum for reviewing a dynamic accident analyzed with a complex systems code.

The overall review included:

(1) identification of the design-basis analyses for postulated accidents and anticipated operational occurrences,

(2) identification of codes, methods, and limitations, based on the team's review of topical reports and NRC Safety Evaluation Reports (SERs), and

(3) an assessment of how limitations, restrictions, and boundary conditions are reflected in the safety analyses.

Central to the assessment was the verification that conditions of approval contained in NRC SERs had been satisfied in the safety analyses. The ISA team also specifically examined code validation using guidance contained in Generic Letter 83-11, "Licensee Qualification for Performing Safety Analyses in Support of Licensing Actions," February 8, 1983.

B.1.3.3 Technical Quality Conclusions

Cycle-specific core performance analyses, such as the CEA drop transient, which used many of the YAEC computer codes and methods, were found to be excellent. An ISA team review of predicted and measured fuel bundle power distributions showed excellent agreement over several fuel cycles. Fuel performance calculations considered the multiple fuel types and multiple projected burnup histories, and were found to be excellent overall. More complicated, less frequently performed systems safety analyses were found to contain weaknesses, such as those associated with the main steam line rupture (MSLR) accident, but the analyses were found to be acceptable based on compensating margin. Overall, the use of analytic codes for safety analyses was evaluated as very good.

B.1.3.4 Compliance with Safety Evaluation Reports Conditions

Compliance with conditions imposed on the use of analytic codes was verified for each of the 67 SER conditions affecting 13 codes. Although full compliance was

confirmed, an audit trail with process documentation to assure compliance was not always available, thereby requiring in some cases additional analyses to verify compliance.

B.1.3.5 Process Conclusions

YAEC did not have a written process to document how safety analyses conformed to code SER conditions. Also, YAEC did not have a documented process in place to identify and rank key phenomena for each of the transients and accidents in the safety analyses report and, in turn, to identify needed code validation and parametric study efforts. During the ISA, YAEC initiated preparation of a "Methods Overview Manual" which was designed to address these process issues.

B.1.3.6 Analytic Code Validation Conclusions

Some codes, such as the physics, fuels, and DNB codes, were found to have extensive validation to actual plant measurements and experimental data respectively. In contrast, the ISA team found that there was an over-reliance on industry RETRAN code validation efforts, and that validation of RETRAN for the MSLR accident was weak. However, the ISA Team determined that regulatory requirements for this validation are not clear. Also, YAEC had submitted and received USNRC approval for using these codes. These submittals included verification of code applicability using validation to previously accepted methods. No validation issues affecting safety were identified.

B.1.3.7 YAEC ISA Response Actions

The ISA review validated the overall technical quality of safety analysis work conducted by YAEC. Several improvement opportunities related to processes and documentation were identified. YAEC addressed these issues through improvements to processes used to validate, control, and document work products. Initiatives included improvements in procedural guidance and requirements, in program oversight (including use of technical specialists), and documentation development and review. These efforts are discussed in the following sections of this Appendix.

B.2 ROOT CAUSE ASSESSMENT

In early 1996, MYAPCo and YAEC executive management also chartered an assessment team consisting of Maine Yankee and YAEC personnel to determine the underlying cause(s) of the allegations. This Assessment Team interviewed over 60 executive, management, supervisory and staff personnel from Maine Yankee and YEAC between February 1996 and April 1996. In April 1996, a report entitled "RELAP5YA Self Assessment" was issued by YAEC, identifying three primary causes for the allegation and nine areas requiring corrective action.

B.2.1 Assessment Process

The assessment team investigated the following programs, processes, and activities:

- Commitment tracking
- Communications with the NRC
- Communications between MYAPCo and YAEC
- MYAPCo oversight of YAEC
- Definition of responsibilities between MYAPCo and YAEC
- 10CFR50.46 reporting process
- Final Safety Analysis Report updating
- YAEC personnel knowledge of NRC regulatory fundamentals
- 10CFR50.59 evaluation process
- Employee Concerns process
- Document control
- Engineering procedures
- Personnel training
- Technical analyses

The assessment team conducted extensive interviews of MYAPCo and YAEC personnel. Interview results were summarized under each category, and evaluated by the assessment team to identify commonalities. The identified commonalities were then evaluated to determine recommendations for improvement or enhancement of apparent weaknesses.

B.2.2 Identified Causes

The assessment team's evaluation identified the following underlying causes:

1. The division of responsibilities/ownership and roles of organizations performing, controlling, administering, and managing activities for the Maine Yankee plant are not always completely and clearly defined or understood by all parties involved in or impacted by the activity.

2. Personnel at YAEC require improved classroom, as opposed to training to the read and sign training approach.

3. Procedures used for controlling the development of analyses should be improved to define user actions rather than having processes driven by personnel knowledge. The current procedures do not require identification of the effects of analyses on licensing commitments or design basis documents.

B.2.3 Assessment Recommendations

In addition to the common causes noted above, the assessment team determined that near-term management attention was necessary for the following areas:

1. Many of the interviewees in support organizations such as Nuclear Engineering, Fire Protection, Licensing, and Technical Support expressed that responsibilities for activities including safety analysis report updates, commitment tracking, interpretation of NRC regulations, interface with the NRC, and distribution of MYAPCo documents are not clearly understood. MYAPCo and YAEC management should define the responsibilities of each organization and clarify their relationship so that programs and processes can function more efficiently.

2. YAEC should improve the controlling procedure for preparation and issuance of analyses to ensure that; (1) regulatory limitations and thresholds are considered upon completion of calculations, (2) licensing design basis documents affected by analyses are changed when needed, (3) use of computer codes are within the NRC's Safety Evaluation Report (SER) bounds for which they were approved, (4) measures are in place for the use, control and dissemination of preliminary data, and (5) procedures appropriately address QA program requirements.

3. The YAEC Employee Concerns Program should be evaluated to ensure that personnel are fully aware of the various mechanisms available to resolve an issue before it becomes an employee concern.

4. Training programs need to be improved for technical personnel. Areas of importance include the relationship and hierarchy of pertinent codes, standards and regulatory requirements to licensing and design basis documents; and procedural training.

5. A commitment tracking system that interfaces with both Maine Yankee and YAEC needs to be developed to ensure that both formal and informal commitments are tracked and accounted for in both organizations.

6. The licensing functions at MYAPCo and YAEC need to be strengthened using appropriate, sufficiently trained personnel who are knowledgeable in all aspects of licensing commitments.

7. YAEC should be included in communications between MYAPCo and the NRC when YAEC supported work is involved.

8. YAEC needs to strengthen the content of its deliverables to include key assumptions, values, and scope to support appropriate use by the receiver and user of the deliverable.

9. Documentation of important communications among the NRC, MYAPCo and YAEC (i.e., mutual understandings or agreements) should be strengthened to assure that commitments and agreements are captured and understood by all parties.

B.3 INITIAL CORRECTIVE ACTIONS

Subsequent to the issuance of the assessment team's report, YAEC developed and implemented a plan to address each of the specific recommendations. These actions were:

1. Relationship between Maine Yankee and YAEC was clarified.

   Engineering for Maine Yankee was reorganized such that the engineering function was controlled directly by a Maine Yankee Engineering Manager and the communication for YAEC support was identified to be through a YAEC engineering coordinator. In the past engineering functions were controlled by a YAEC Project and Engineering Manager. The licensing function was totally at Maine Yankee with support from YAEC individuals who reported to Maine Yankee management.

2. Five (5) NED procedures were developed in the following areas:

   - Analysis/calculation review checklist
   - Safety and relief valve monitoring
   - 10CFR50.46 reporting
   - Safety analysis
   - Changes to licensed methods

   These procedures provided additional guidance for NED personnel who were involved in the development and review of engineering analysis. The procedures described attributes that engineers used to assure that analyses met the administrative and technical requirements of the higher level procedures controlling design activities. They also clarified the steps for analyzing results to determine when results of analysis could potentially be reportable, especially under the requirements of 10CFR50.46.

3. Employee Concerns Program was revised.

   YAEC formed a task force to review the method whereby reporting employee concerns were identified. As a result of this task force, the procedure was modified to make personnel aware of the various methods for bringing a concern to the attention of management without recourse to the individual. YAEC also established a twenty-four hour hotline that permits anonymous reporting of concerns. The hotline also permits individuals to obtain the status of actions taken to resolve a concern

4. Formal training on codes, standards, and regulatory requirements was provided to key personnel.

Individuals received training from professional organizations on the codes and standards applicable to the work in which they are involved. YAEC also committed to formal classroom training for all engineering personnel on the requirements of the Engineering Instruction Manual. The goal was to lower the incidence of "failure to follow procedures" and to increase the quality of engineering output documents.

5. Commitment tracking systems were upgraded.

YAEC engineering functional areas developed tracking systems that allowed management to assign individuals to particular tasks and also to prioritize each task according to their importance and need.

6. YAEC supported the increase and reorganization of the Maine Yankee licensing staff.

In 1996 the licensing function at Maine Yankee was reorganized and expanded so that it could better perform the licensing activities with the NRC. Maine Yankee licensing staff was increased to handle certain support work that was formerly done at YAEC. Although YAEC continued to provide some licensing support to Maine Yankee, it is clearly reinforced that the full licensing function for Maine Yankee resided with MYAPCo.

Also identified were issues requiring further action by both MYAPCo and YAEC. A YAEC QA surveillance was conducted to verify the actions taken in response to the assessment team's report. The surveillance concluded that actions had been undertaken to resolve the identified issues and that a tracking system had been developed to identify actions, assign responsible individuals, and monitor the status of action completion.

In June 1997, YAEC QA conducted a performance-based audit of NED safety-related engineering activities with the assistance of Scientech as independent technical specialists. The focus of this audit was to perform a full scope review of the fuel reload analyses, as well as administrative issues including self assessment and corrective action. The Scientech team report concluded that:

> ".....the technical basis ..... appeared sound. Only minor deficiencies were identified in the calculational notebooks reviewed."

The audit also included an examination of sixteen YAEC self-assessments and one functional area assessment conducted by NED. The audit concluded that NED was:

> "..... effective in identifying problems and areas for improvement and recommending corrective actions."

B.4 PROCESS IMPROVEMENTS

YAEC recognized that improvements needed to be made in the methods for controlling work activities, training personnel, enhancing the design control procedures and increasing personnel awareness for reporting and resolving issues. During the latter part of 1996, a plan had been established for these improvements. A brief summary of the key changes made at YAEC is provided in this section:

B.4.1 Corrective Action Process

B.4.1.1 History of YAEC Deficiency Reporting Systems

The YAEC Quality Assurance Program Manual includes a commitment to ANSI N45.2.11 and NRC Regulatory Guide 1.64, both of which describe the Quality Assurance requirements for design control applied to nuclear power plants. ANSI N45.2.11 requires that procedures be established for reporting deficiencies [in design documents] and corrective action to appropriate levels of supervision and management. Furthermore, the standard requires that a cause be determined for significant or recurring deficiencies or errors and that changes be made in the design process to prevent them for occurring again.

In order to comply with the standard's requirement, YAEC established an engineering instruction, "Engineering Deficiency Report (EDR)," which described the methods for reporting design deficiencies within all YAEC projects and engineering functions. The procedure was basic. It included a section to describe the deficiency, a section to describe the corrective and preventive action, and a section to indicate completion and verification of the corrective and preventive action. The procedure was in effect from 1989 to the end of 1997.

A second deficiency reporting system was the Status and Summary of Corrective Action (SSCA) report. This report was instituted in the 1970s to report findings identified during audits. This was in response to the requirements of ANSI N45.2.12.

B.4.1.2 Condition Report System

Each of the systems described above provided different methods for reporting deficiencies. However, each system had common problems in regard to in the level of management attention afforded to monitor performance on a company-wide basis and to prevent recurrence of problems. In addition, neither system had been designed to collectively analyze performance data on a periodic basis for determining adverse trends within the company.

On May 31, 1997 YAEC issued Technical Administrative Guideline No.25, Condition Report (CR) System as a replacement for the previous deficiency reporting systems. The reason for issuance of the single reporting system was to assure a consistent method for reporting, evaluating and dispositioning deficiencies. The system was also designed to provide a much lower threshold for reporting deficiencies. Significant improvements resulted from the CR process and are summarized as follows:

1. The system allows personnel to not only identify actual deficient problems in violation of a requirement but also potential problems. This created a system that could be an integral part of the self-assessment process.

2. The system requires that all conditions reported under a CR be evaluated by the CR Review Committee.

3. All CRs are evaluated by the individual and the individual's supervisor prior to corrective action evaluation to determine if the situation could be potentially reportable to any local, state, federal or other regulatory agency. In addition, each of the CRs are evaluated to determine if they directly impact one or more plants and whether the adverse condition could potentially impact plant operations. If so, the CR system has provisions for reporting potential operability concerns directly and immediately to the plant.

4. The system established a Condition Report Review Committee (CRRC) which has representatives from the Executive Office, Quality Assurance, Environmental, Licensing, and Nuclear Engineering functional areas. The CRRC is responsible for the review of all CRs. The CRRC review requires evaluation of any specified corrective and preventive actions, assignment of corrective and preventive actions and action parties, determination of the appropriateness of reportability and operability, assignment of the significance of the CR, and evaluation of the need for a root cause analysis and follow up verification of corrective actions.

5. The CR system does not allow extensions for providing responses or for completing corrective actions. The system tracks the age of both responses (plans) and actions, and thus heightens the monitoring and awareness of overdue items by executive level and line management.

6. The CR system includes a uniform method of coding problems so that they can be trended periodically and analyzed for adverse process conditions which may not be apparent from examination of isolated events.

7. The CRs are controlled in a single tracking system and the status of open items is available to all personnel at all times through an integral, company-wide electronic network system.

YAEC personnel were formally trained on the use of the CR system and were given the opportunity to enhance the procedure prior to its formal issuance. In summary, the CR system provides a single problem reporting system with a lowered threshold for reporting problems and it receives the attention of executive and line management for every situation reported.

B.4.1.3 Deficiency Trends

EDRs, SSCAs and CRs generated from 1995 to 1997 were evaluated to identify trends in deficiencies and to examine the effects of the recent charges to the deficiency report system.

In September 1997, the first trend report from the newly established Condition Report System was issued.

Within these categories, it was difficult to recognize any type of trend up to this point, since the system had been in place for four months. However, it was apparent that the threshold for initiating a deficiency report had been lowered significantly. The total number of various YAEC deficiency reports issued during the 1995 to 1997 time frame is shown below.

Year	EDRs	SSCAs	ERs[1]	CRs	Total
1995	21	27	-	-	98
1996	45	66	18	-	129
1997	37	32[2]	-	199	268

[1]) Event Reports were issued to Vermont Yankee in lieu of SSCAs.
[2]) Includes 29 recommendations which were not issued as an SSCA.

Thirty-two percent (32%) more deficiencies were reported in 1996 than in 1995 and in 1997 there were one hundred-eight (108%) percent more deficiencies reported than 1996. This development was directly attributed to a lower threshold, more self-assessments being performed, and an increase in a questioning attitude by employees.

B.4.2 Employee Concerns Program

The YAEC Employee Concerns Program was revised in 1997. The Employee Concerns Program now includes:

- An employee hotline that is available twenty-four hours per day, seven days a week. The hotline does not require identification of the individual and yet is capable of being queried by a concern's originator to determine the resolution status of a concern.

- Assurance of confidentiality for individuals reporting a concern.

- Assurance that responses are provided to all concerns.

- Provisions for ensuring personnel are aware of the various methods for reporting concerns, both internally and externally.

B.4.3 Engineering Instructions

The Engineering Instructions (WEs) were audited a number of times during the 1995 to 1997 period. The need for improvement was stressed during interviews conducted during the RELAP5YA Self Assessment. Although procedures met the requirements of ANSI N45.2.11, NQA-1, and Regulatory Guide 1.64, several major changes were made to the procedures to ensure that:

- Calculations and analyses included a detailed review of uncertainties,

- Results of calculations and analyses are reviewed against allowables to determine if the results need to be reported to any regulatory agencies,

- Preliminary results are so designated when transmitted and that they are followed up with approved calculations and compared to preliminary results,

- Records are established to provide configuration control so that potential revisions can be evaluated against previous results,

- Requirements are established to verify that contractors receive training on applicable YAEC procedures prior to starting work,

- Commercial software is verified on a case by case basis,

- Objectives of the analysis are clearly defined and determined to be met,

- Analyses clearly specify the end use and restrictions on the use of analysis results, and

- Independent reviewer comments include a detailed description of the review and a detailed listing of all comments generated during the review of the analysis for documented dispositioning with the preparer.

YAEC Design Control Measures were evaluated and assessed. The purpose of this review was to compare and contrast YAEC's Procedures with respect to ANSI N45.2.11 (1974) (endorsed by Regulatory Guide 1.64, Revision 2) and ASME NQA-1-1989 through the NQA-1b-1991 Addenda).

Specifically, YAEC's Procedures were reviewed to establish that adequate and sufficient controls exist and are documented for the following Design Control Criteria:

- Program Requirements
- Design Input Requirements
- Design Process (Design Analysis, Drawings and Specifications)
- Interface Control (Internal & External)
- Design Verification
- Document Control
- Design Change Controls
- Corrective Action
- Records
- Audits (Internal & External)
- Software Quality Assurance

Based on this assessment it was determined that YAEC's implementing procedures, as defined in various Engineering Instructions, Technical Administrative Guidelines and Quality Assurance Procedures, provide adequate and sufficient controls in the area of Design Control. Further, these implementing procedures satisfy the requirements of:

- ANSI N45.2.11 (1974)
- Regulatory Guide 1.64, Revision 2
- ASME NQA-1-1989 through the NQA-1b-1991 Addenda

In addition, the most recent changes to these Engineering Instruction, as documented in Engineering Modification Nos. 50 and 51, were reviewed. The Engineering Modifications provided a descriptive summary of the changes along with the reasons for the changes. In all instances, changes to the Engineering Instructions were viewed as enhancements to the existing Design Control measures.

B.4.4 Joint Quality Audit Group

In 1996, YAEC QA Department formed an organization named the Joint Quality Audit Group (JQAG) whose mission was to provide plant affiliate oversight of the YAEC engineering activities. The JQAG is comprised of QA representatives from, YAEC, Maine Yankee, Vermont Yankee, Seabrook, Northeast Utilities and Boston Edison. The JQAG meets quarterly and reviews oversight activities performed and planned. The basic mission of the JQAG has been designed as follows:

1. Discuss YAEC audit and surveillance issues or concerns which are of common interest or which may have potential impact upon YAEC QA Department products and services;

2. Discuss YAEC QA Department Functional Area Assessment results including action plans and updates to the plans;

3. Facilitate discussion of emerging industry issues and share available information from industry organizations, such as NRC, NEI, and EPRI;

4. Review and comment on the YAEC-QAS generated audit schedules for internal and plant audits;

5. Facilitate participation on selected YAEC audits;

6. Report on the status of the YAEC audit program, upon request, to: (I) Nuclear Safety Review Committee (NSARC) of plants receiving YAEC technical services, (ii) YAEC Board of Directors and (iii) other appropriate requestors.

7. Provide input to plans for an independent audit of YAEC.

8. Provide a forum for establishing a common audit program which can be used at each of the JQAG member plants. This will include the development of procedures, schedules, reporting techniques and consistent follow-up of open items.

B.4.5 Functional Area Representatives

In order to provide more frequent and consistent internal oversight of engineering activities, YAEC QAS established QA functional area representatives. The representatives are responsible for day to day contact with the applicable functional area, performing surveillances of functional areas, reviewing procedures and assisting in resolving quality matters. In 1996 and 1997, the QA representatives performed a total of twenty-three (23) surveillances which supplemented the regularly scheduled audits.

B.4.6 Technical Specialist Program

An initiative was undertaken in 1996 to increase the use of technical specialists on the internal audits. During the period from 1995 to 1996, there were a total of thirty internal audits performed and there were technical specialists from Northeast Utilities, Scientech, YAEC, Boston Edison, Seabrook, Vermont Yankee, South Carolina Electric & Gas, New York Power Authority, and also independent consultants to assist YAEC QA. The use of additional technical specialists allowed for more in-depth verification of technical matters and also for more objectiveness in reporting results.

B.4.7 Functional Area Assessments

In 1996 and 1997, YAEC departments were responsible for preparing a Functional Area Assessment (FAA). This process required the YAEC departments to evaluate their own performance and grade themselves according to the results of audits, self assessments, inspection reports, surveillances and management expectations. The reports included a description of the tasks performed by the functional areas, a list of department strengths and weaknesses, and an improvement plan to address the weaknesses in the upcoming year. In addition, Vermont Yankee had set client expectations to implement FAAs in YAEC departments which perform a significant amount of work for them. The FAA was used by the QA Department to determine the amount of oversight required in each functional area and also to determine generically the focus for every audit.

B.4.8 Self Assessments

In 1995, few YAEC departments were performing self assessments. This was mainly due to the fact that management expectations were not communicated to all levels of the organization. This became widely known after an audit was performed in early 1996. As a result of that audit, department management began performing self assessments and staff personnel became more aware of the benefits of performing them. The process for documenting, tracking issues and corrective actions was inconsistent throughout the company. In 1997, self assessments were being performed by all departments on a more routine basis. These assessments were value added, meaningful and very insightful as compared to the ones performed during 1995 and 1996. Departments were tracking issues and correcting them, they were being brought to the attention of management, and in some departments, benchmarking and performance goals were being used to evaluate their performance. A shortcoming of the process was that action items were not being tracked in a single system. This shortcoming is to be resolved in 1998.

B.5 AUDITS AND ASSESSMENTS

Throughout 1995 to 1997, YAEC engineering activities were subjected to numerous audits by YAEC QA and independent assessments by outside organizations. Each YAEC organization

involved in safety-related activities was subjected to audits and/or assessments. The results of these activities are summarized in the following subsections.

B.5.1 Internal Audits/assessments of Engineering-Related Activities

Twenty-five (25) audits were performed by YAEC QA of engineering activities performed by YAEC project groups. Of these 25 audits, two (2) multi-department audits were conducted in 1997, rather than the individual department audits as performed in 1995 and 1996.These audits and assessments were both scheduled and unscheduled, addressing both routine QA audits and special audits in areas identified by YAEC management. Frequently, the audit teams included technical specialists. These specialists were specifically matched to the engineering disciplines which were the subject of the audits Specialists were obtained from: (I) within YAEC, (ii) plants inside and outside NRC Region I, and (iii) independent consulting firms. This use of technical specialists provides a valuable blending of technical and programmatic expertise. Additionally, the knowledge and experience of technical specialists outside of YAEC allows the engineering activities being assessed to be benchmarked against non-YAEC organizations. Common areas requiring improvement were found in the areas of: (I) program ownership, (ii) analyses inconsistencies, (iii) documentation control, (iv) department procedure enhancements, and (v) software control.

B.5.1.1 Ownership of Programs

During various project audits, identified issues indicated that (I) interfaces between YAEC projects and their respective plants and (ii) ownership of various technical programs (e.g., Appendix R, EQ, design control) were weak. Actions were developed to address these weaknesses. As a result, clear and concise procedures were implemented and project meetings with responsible plant personnel were held to resolve and strengthen lines of communication and ownership.

B.5.1.2 Adherence to Procedural/Code Requirements

A number of identified weaknesses were "failure to follow" procedures/code requirements. The results indicated that engineering instructions governing implementation of code requirements were not consistently being followed. Training sessions were established on the YAEC Engineering Instructions (WEs) and personnel accountability for procedural compliance was stressed. NED checklists were implemented as part of calculation closure to ensure administrative compliance with the WEs.

Additionally, a task force was established to review the inconsistencies in the WEs and to revise them, as necessary. As a result of this effort, (I) procedures

controlling the preparation and review of analyses are clear and concise for the user, and (ii) WEs interface more effectively with each other.

B.5.1.3 Analysis Inconsistencies

Issues identified during audits indicated inconsistencies in analyses. These inconsistencies were brought to the attention of project management. Technical specialists utilized as members of the engineering audit teams noted that although analysis inconsistencies were identified, no circumstances were found where they affected the operability of the respective plants. However, a small percentage of them did result in the generation of either a Basis for Maintaining Operation (BMO) or a Justification for Continued Operation (JCO) document. Corrective actions focused on stronger design reviews, more descriptive assumptions and inputs within the analyses, and more thorough/detailed comment resolution of independent reviewer comments.

B.5.1.4 Control of Documentation/Records

Audits identified documentation inconsistencies and record turnover issues. Areas for controlling document turnover as a QA record to document control were reviewed in audits of the project groups at YAEC. Adherence to ANSI N45.2.9 in the past was confusing because of the general guidance provided for record storage. Corrective actions taken by the YAEC project groups included several task forces with Document Control personnel and the QA Manual revisions. Analysis procedures now contain detailed instructions for record control, addressing both temporary and permanent storage requirements.

B.5.1.5 Software QA Control

Assessments indicated that software generated by engineers outside of the Software Control Library did not consistently meet engineering instruction requirements. Corrective actions generated by project groups for audit issues led to the revision of the engineering instructions for software control. The engineering instructions for calculation control was revised to provide better guidelines in the control of various computer codes generated. Revised software control procedures are more detailed and user friendly. Additionally, enhancements have been made to (I) the Software Control Library and its implementing procedure, (ii) the interface between the software control and calculation procedures, and (iii) the control of software used by project engineers.

B.5.2 Joint Utility Management Audits

10CFR50 Appendix B, ANSI N45.2 and ANSI N18.7 require that all aspects of the QA Program be audited, and also that the adequacy and effectiveness of the QA program be regularly assessed. In addition, plant technical specifications require that the QA programs be audited at least every two years. Since YAEC QA supplied QA services to many Yankee plants, YAEC QA became a member of an organization called the Joint Utility Management Audit (JUMA) group in order to fulfill these NRC requirements and to remain independent for this activity. JUMA performs audits and assessments of member utilities with representatives independent of a specific utility. As an example, representatives from utilities A, B, and C would audit utility D, thus providing the independence and objective overview of QA program implementation required by federal regulations and industry standards. There were three JUMA audits performed at YAEC during the period of 1995-1996.

B.5.2.1 1995 JUMA Audit of Vermont Yankee

The 1995 audit consisted of a team with representatives from Philadelphia Electric Company (PECO), South Carolina Gas & Electric (SCE&G) and Commonwealth Edison (ComEd). The focus of the audit was YAEC functions for the Vermont Yankee plant QA program; the specific elements reviewed during the audit were as follows:

- Corrective Action /Audits
- Peer Inspection / Audits / Surveillances
- Vendor Quality Assurance

The audit of corrective action activities included a review of the adequacy of management support for the new Vermont Yankee corrective action system, QA involvement in the system, use of the system during audits, and threshold for reporting deficiencies. The auditors interviewed personnel and evaluated a sample of audit reports with findings, and a sample of corrective action documents. As a result of assessing this area, the audit team concluded that the Vermont Yankee corrective action system was effective and provided enhanced problem reporting, increased management awareness of problems, and simplified process control.

The second area assessed by the JUMA team, peer inspection, included an assessment of the adequacy of the QA audits and surveillances over peer inspection activities, adequacy of QA reviews of contractor QC, and effectiveness of attributes selected for peer inspection. The report concluded that audits and surveillances over peer inspection activities were adequate, but could be improved by ensuring aggressive corrective action was taken by audited organizations. In addition QA involvement in and oversight of

contractor QC activities could be increased. Attributes used to perform inspections were consistent with procedures and regulatory requirements.

The last area assessed during the 1995 JUMA audit was the Vendor QA activities. The specific elements reviewed under this area included:

- Procurement Engineering and Vendor QA redundancy
- Communications between the Plant Maintenance, Procurement Engineering, and Vendor QA
- Vendor QA Oversight of Receipt Inspection

The audit concluded that the Procurement Engineering (PE) and Vendor QA (VQA) do not duplicate their activities. Communications among each of the involved procurement organizations is satisfactory, for the most part, although some improvements could be made between VQA and PE. Vendor QA oversight of receipt inspection was considered satisfactory.

The results of the audit did not identify any deficiencies in program implementation. There were ten (10) recommendations for improvements in the audited areas which were addressed and resolved by both Vermont Yankee and YAEC.

B.5.2.2 1996 JUMA Audit of Maine Yankee

The 1996 JUMA audit was conducted of Maine Yankee QPD by a team consisting of Florida Power & Light, Entergy, and American Electric Power representatives. The scope of the audit included the in-plant audit program, the YAEC internal audit program and the plant QC inspection program.

The evaluation of the in plant audit program was accomplished by interviewing audit personnel and customers of the audit program, review of audit and surveillance documentation including reports, procedures and corrective action documentation. The report identified that the audit program was being effectively implemented and that the personnel conducting the audits had good knowledge of the areas being audited. The audit process made good use of technical specialists and the reports were well written. The audits that were reviewed were in compliance with the QA program and technical specifications.

The receipt and in-plant QC programs at MY were reviewed. The assessment included an examination of documentation, observation of activities and interviews with personnel. The audit team concluded that the QC program was effectively implemented and generally meets the needs of the plant.

Opportunities for improvement included better oversight of receipt inspection, and more use of analytical equipment to verify material properties

The last area assessed by the 1996 JUMA audit team was the YAEC internal audits of functional areas providing services to Maine Yankee. The evaluation included a review of audit schedules, plans, reports, corrective action documents, and interviews with selected personnel. Based on the reviews and interviews, the audit team concluded that the audits were very thorough and effective. Everyone on the audit team believed that the audits were more effective now than those performed five years earlier. They did, however, make several recommendations for improvement. First, they recommended that the audit group track the 10CFR50, Appendix B, elements and functional tasks audited to assure comprehensive coverage, and also to obtain input from audited organizations for activities to be considered during the audit. Lastly they recommended that audits increase the use of technical specialists from outside the region to facilitate bench marking. The audit team further identified a weakness with the oversight of YAEC audit activities by Maine Yankee. They encouraged their participation in the newly formed Joint Quality Audit Group (JQAG) which was established to provide the sponsor plants with the opportunity to input into YAEC audit plans, participate on audits, and to recommend corrective actions for future audits.

The 1996 JUMA audit did not identify any deficiencies; however, their were thirteen recommendations for improvement. Only three the recommendations noted above pertained to YAEC. Subsequent to the audit, YAEC QA acted upon the JUMA recommendations by developing a comprehensive matrix identifying areas and elements audited, by requesting customer input to audit plans, and by utilizing technical specialists from outside the region to the maximum extent possible.

B.5.2.3 1997 JUMA Audit of Vermont Yankee

The 1997 JUMA audit focused on the Vermont Yankee plant and assessed the corrective action programs, the audit program and vendor QA activities. The audit team consisted of representatives from ENTERGY, South Carolina Electric & Gas, PECO Energy and a peer evaluator from Vermont Yankee.

The evaluation of the corrective action system included a review of the newly instituted Condition Report (CR) system at YAEC. The audit team reviewed the procedure, a sample of reports, and interviews with personnel. The team concluded that the CR process has high level management involvement and that the QA personnel involved in the process displayed a lot of dedication and ownership. Their was some concern raised about the timeliness of instituting procedural changes for the system, and also for the backlog of overdue action

items. The QA department has since made improvements in the CR system based on the recommendations from JUMA. Less than a month after their audit, the CR procedure was revised to reflect the current practices for process control of the CR. Backlogged issues were brought to the attention of YAEC management and the overdue items dramatically reduced.

The audit team also evaluated the internal YAEC audit program by reviewing audit schedules, plans and reports. The team concluded that the audits were of sufficient depth and scope, and that they were performed by personnel who had enough technical experience to identify safety significant issues. The team also stated that the use of QA functional area representatives has improved communications between departments. Lastly, the team identified that the auditors utilized performance based information obtained from Functional Area Assessments (FAAs) to further explore weaknesses and strengths during audits. There were four recommendations relative to improving the audit process. The first was to become more aggressive in assuring the timely resolution of issues. The second recommended that management provide better direction for the performance of self assessments. The third recommendation pertained to revising the surveillance and audit procedures so that they remained current with actual practices. Lastly, they recommended that the audit group review audit plans against the most recent FAA to assure adequate oversight. The recommendations were acted on by appropriate management and resolved prior to end of 1997.

The JUMA team also reviewed the vendor QA activities related to the Vermont Yankee Cycle 20 core design and reload analysis. The review specifically included an evaluation of the organizational interfaces, oversight by QA and Fuels departments, and exchange of key design documents. The JUMA auditors concluded that vendor QA activities were effective. There were four recommendations for improvement in the areas of developing a reload interface procedure, establishment of a single point interface contact, development of an emerging issue action plan, and changing the time requirement for issuance of vendor audit and surveillance reports. Each of the recommendations in the vendor QA area have been addressed by QA management and resolved.

The conclusion reached by the 1997 JUMA audit team was that areas evaluated were adequate and effective. There were no deficiencies identified, however there were fourteen recommendations for improvement, all of which were addressed by QA management.

APPENDIX B

REFERENCES

B.1 Letter, G. D. Whittier (MYAPCo) to USNRC, MN-96-08, "Docket No. 50-309, "Maine Yankee Atomic Power Station Response Team and Independent Review Team Reports," dated January 22, 1996.

B.2 R. T. Fernandez, et al., "RELAP5YA - A Computer Program for LWR Thermal-Hydraulic Analysis," Report YAEC-1300P, Volumes 1, 2, and 3, October 1982.

B.3 Letter, USNRC to Maine Yankee, "Acceptance for Referencing Topical Report YAEC-1300P, Volumes 1, 2, 3, RELAP5YA: A Computer Program for Light Water Reactor System Thermal-Hydraulic Analysis," dated January 30, 1989, with enclosures.

B.4 Ransom, V. H., et al., "RELAP5/MOD1 Code Manual, Volumes 1 and 2, NUREG/CR-1826, March 1982.

B.5 Letter, L. H. Heider (YAEC) to D. G. Eisenhut (USNRC), FYR 83-9, FVY 83-4 and MN 83-12, dated January 14, 1983.

B.6 Willcutt, G. J. E., Jr., Letter Q-7-83-559, "Review of Yankee Atomic RELAP5YA Small Break LOCA Model," Los Alamos National Laboratory, dated November 17, 1983 (from Reference NMY 86-51).

B.7 Fineman, C. P., "Technical Evaluation Report: RELAP5YA Computer Program for Use in PWR Small Break Analysis", Idaho National Engineering Laboratory, EGG-TFM-7933, dated May 1988.

B.8 Letter, G. D. Whittier (MYAPC) to D. G. Eisenhut (USNRC), MN-84-56, "Resolution of TMI Action Item II.K.3.5, 'Automatic Trip of Reactor Coolant Pumps'," dated April 9, 1984.

B.9 Letter, USNRC to Maine Yankee, "Maine Yankee Reactor Coolant Pump Trip," dated April 15, 1986, NMY 86-51.

B.10 YAEC-1868, "Maine Yankee Small Break LOCA Analyses," 1993.

B.11 Letter, James R. Hebert (MYAPCo) to W. T. Russell (USNRC), MN-96-018, dated February 21, 1996.

B.12 Letter, Patrick M. Sears (USNRC) to C. D. Frizzle (MYAPC), "Maine Yankee: Implementation of Small Break LOCA Analysis, NUREG-0737 II.K.3.30 and II.K.3.31 (TAC 48176), dated May 8, 1989.

B.13 Letter, S. P. Schultz (YAEC) to E. H. Trottier (USNRC), "Information Regarding the Small Break Loss of Coolant Accident (SBLOCA) Analysis for Cycle 15 Operation at Maine Yankee," dated December 14, 1995.

B.14 Letter, C. D. Frizzle (MYAPCo) to W. T. Russell (USNRC), MN-96-056, "Submittal of Maine Yankee SBLOCA Licensing Analysis in Compliance with 10CFR50.46 and in Satisfaction of TMI Action Items II.K.3.30, II.K.3.31, and II.K.3.5," dated April 25, 1996

APPENDIX C

RELAP5YA SBLOCA TECHNICAL ISSUES ASSESSMENT

DUKE ENERGY CORPORATION
DUKE ENGINEERING & SERVICES, INC.

YANKEE ATOMIC SMALL BREAK LOCA
TECHNICAL REVIEW REPORT

Technical Review Conducted For:

Duke Engineering & Services, Inc.

Location: Duke Engineering & Services Inc. - Bolton, Massachusetts

Scope: Application of RELAP5YA in Small Break LOCA Analyses at Maine Yankee

Dates: February 2 through February 6, 1998

PERFORMED BY:	Z. L. Taylor	Duke Power Company
	G. B. Swindlehurst	Duke Power Company
	B. M. Dunn	Framatome Technologies
	L. E. Hochreiter	Pennsylvania State University

PREPARED BY: G. B. Swindlehurst **DATE:** February 25, 1998

G. B. Swindlehurst

Z. L. Taylor **DATE:** February 25, 1998

Z. L. Taylor

APPENDIX C

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I. EXECUTIVE SUMMARY

During the period of February 2 through February 6, 1998, a Duke Engineering & Services, Inc. (DE&S) sponsored technical review of activities at the DE&S (formerly Yankee Atomic) Bolton office was conducted by a team composed of industry experts in the area of LOCA analysis. It was the purpose of this review to provide an independent technical review of Yankee Atomic's small-break LOCA evaluation model (RELAP5YA SBLOCA) as applied to Maine Yankee and to determine compliance with 10CFR50.46 and 10CFR50 Appendix K requirements. The Review Team also evaluated Yankee Atomic's cognizance of the potential technical issues with the model, adequacy of corrective actions that may have been required, and effectiveness of communication of these with Maine Yankee and the Nuclear Regulatory Commission.

During the course of the review numerous technical documents and memoranda were examined. In addition, interviews were conducted with key Yankee Atomic staff members involved in the development of RELAP5YA and its application to Maine Yankee. Overall, based on the information available to the Review Team, including other analysis results, it was concluded that the results of Yankee Atomic's SBLOCA calculations for Maine Yankee are consistent with results of some other calculations by other codes and organizations. However, Yankee Atomic failed to fully appreciate NRC expectations, as understood within the nuclear industry's LOCA community, for 10CFR50.46 model acceptance. Although these expectations are not completely documented in specific references, such as the NRC Standard Review Plan or Regulatory Guides, they have been clearly communicated to the industry's LOCA community through extensive and numerous interactions since the Interim Acceptance Criteria of 1971. The interactions have made it clear to what degree the community was to keep the NRC informed, what types of issues needed NRC approval, and what the NRC expectations were for approved evaluation models. Yankee Atomic would have benefited from the experience of companies or individuals who more fully participated in the process. The Review Team recognizes that it remains Yankee Atomic's responsibility to understand NRC's expectations. Yankee Atomic would also have benefited from improved, continuous and direct technical communications with the NRC Reactor Systems Branch, which could have eliminated many of the issues of concern as identified by the Review Team.

Overall observations and conclusions of the team are summarized as follows:

1. There was a failure of the Yankee Atomic organization as well as the LOCA Group in fully understanding NRC expectations (in particular the NRC's Reactor Systems Branch) for an Appendix K evaluation model, as understood within the industry LOCA community, for 10CFR50.46 model acceptance. Plant specific calculations were undertaken too late to change the direction of the development project following the identification of significant code limitations.

2. During the development of the Maine Yankee specific SBLOCA model and analyses there were a number of organizational barriers and a culture which impeded the LOCA Group from completing the project in a manner which met NRC expectations.

3. The Yankee Atomic organization was aware of performance problems with the RELAP5YA code. These code problems were addressed by the LOCA Group from a technical perspective, however the organization did not adequately maintain communications with the NRC technical staff. This resulted in a failure to resolve all technical issues necessary to satisfy NRC expectations prior to implementation for the Cycle 14 reload.

4. The technical capabilities of the Yankee Atomic LOCA Group are acceptable for performing LOCA analysis. The LOCA Group was qualified and knowledgeable regarding SBLOCA computer models and phenomena. Interviews and documents did not indicate any evidence of deliberate violations of NRC regulations.

5. Based on the information available, including other analysis results, it can be concluded that the results of Yankee Atomic's SBLOCA calculations for Maine Yankee are consistent with results of some other calculations by other codes and organizations. Therefore it is concluded that the SBLOCA PCTs for Maine Yankee meet the 2200°F criterion of 10CFR50.46, and that SBLOCAs remain bounded by LBLOCAs for this plant.

6. Yankee Atomic's direct interface with the NRC on the application of the RELAP5YA model appears to have ended once the SER was received in 1989. Not maintaining a direct interface between the LOCA Group and the NRC technical staff resulted in a loss of communications regarding NRC's expectations for SBLOCA calculations performed with RELAP5YA to support the Maine Yankee license.

7. Continuous communications with the NRC staff could have eliminated many of the issues of concern.

A more detailed discussion on each of the above observations and conclusions is provided in Section III.D of this report.

II. INVESTIGATION

A. List Of Technical Review Team Members

Z. L. Taylor (Team Leader) — Duke Power Company

- 18 years experience with Duke Power Company as Engineering Supervisor and as Licensing Manager at Catawba Nuclear Station
- Currently Nuclear Support Manager in Duke Power's Nuclear Assessment and Issues Division

G. B. Swindlehurst — Duke Power Company

- 20 years at Duke Power Company in Safety Analysis area which includes non-LOCA accident analysis
- Currently Manager of Duke Power's Safety Analysis Group in the Nuclear Engineering Division

B. M. Dunn — Advisory Engineer at Framatome Technologies (formerly B&W)
- 29 years of LOCA analysis experience

L. E. Hochreiter — Pennsylvania State University
- Greater than 25 years experience in LOCA analysis with Westinghouse
- Currently Professor of Nuclear Engineering at Pennsylvania State University

Assisted By:

J. M. McGarry	Attorney at Law with the firm of Winston & Strawn
M. J. Wetterhahn	Attorney at Law with the firm of Winston & Strawn

B. List Of Personnel Contacted During The Investigation

Michael Scott	Engineer	DE&S/YAEC
Steve Schultz	Mgr N&F Serv.	DE&S/YAEC
Bob Harvey	Engineer	DE&S/YAEC
Ramu Sundaram	Engineering Consultant	DE&S/YAEC
Liliane Schor	Project Manager II	DE&S/YAEC
Bruce Slifer	Engineer	DE&S/YAEC
Jim Chapman	Gen Mgr Nuclear Engr	DE&S/YAEC
Mark Robeck	Attorney at Law	Baker & Botts
Glenn Swanbon	Engineer	Northeast Utilities
Pete Anderson	Maine Yankee Interface	DE&S/YAEC
Mike Lennon	Attorney at Law	Baker & Botts

C. Entrance Meeting And Exit Debriefing

Entrance Meeting: February 2, 1998

Z. L. Taylor	Nuclear Support Mgr NAID	Duke Power Co.
G. B. Swindlehurst	Mgr Safety Analysis	Duke Power Co.
J. M. McGarry	Attorney	Winston & Strawn
M. J. Wetterhahn	Attorney	Winston & Strawn
L. E. Hochreiter	Professor of Nuclear Engr.	Penn State University
B M. Dunn	Advisory Engineer	Framatome Technologies
Greg Hudson	Project Manager	DE&S

Jim Chapman	Gen Mgr Nuclear Engr	DE&S/YAEC
Paul Bergeron	Mgr THSAG	DE&S/YAEC
Steve Schultz	Mgr N&F Serv.	DE&S/YAEC
Bob Harvey	Engineer	DE&S/YAEC
Chris Hickey	Senior QA Engr	BECo.
Liliane Schor	Project Manager II	DE&S/YAEC
Ramu Sundaram	Engineering Consultant	DE&S/YAEC
C. D. Nunzio	Senior QA Engr	DE&S/YAEC
Wayne Merritt	Project Mgr	DE&S/YAEC
Mike Lennon	Attorney	Baker & Botts (Houston)
Mike Scott	Engineer	DE&S/YAEC

Exit Debriefing: February 6, 1998

Z. L. Taylor	Nuclear Support Mgr NAID	Duke Power Co.
G. B. Swindlehurst	Mgr Safety Analysis	Duke Power Co.
Greg Hudson	Project Manager	DE&S
Steve Schultz	Mgr N&F Serv.	DE&S/YAEC
Pete Richardson	Independent Ass	NAES Co.
Jim Chapman	Gen Mgr Nuclear	DE&S/YAEC
Mike Scott	Engineer	DE&S/YAEC
Frank Sabadin	QA	DE&S/YAEC
Bob Harvey	Engineer	DE&S/YAEC
Paul Bergeron	Mgr THSAG	DE&S/YAEC
Rich Faix	Engr. Supervisor	NAES Co.
Mark Robeck	Attorney	Baker & Botts
Liliane Schor	Project Manager II	DE&S/YAEC
Wayne Merritt	Project Manager	DE&S/YAEC
Camillo A. DiNunzio	Sr. QA Engr.	DE&S/YAEC
Steve C. White	Ops QA Mgr.	DE&S/YAEC

D. Technical Review Process

Copies of technical documents were reviewed prior to the team's arrival at the DE&S Bolton office.

An entrance meeting was held with appropriate DE&S management to discuss the technical review plan for the week. Refer to Section C of the report for list of attendees.

Interviews were conducted with key personnel involved in the development of the RELAP5YA SB LOCA evaluation model, its application to Maine Yankee, and the best estimate approach for addressing NUREG-0737 and related 10CFR50.59 issues. These interviews were held with all levels of the Nuclear Engineering Department organization in place at the time of this work, including director level, LOCA Group managers, lead

engineers, as well as other engineers involved with the technical aspects of the work. Technical work reviewed included (but was not limited to):

- The RELAP5YA SBLOCA (YAEC-1300P)
- The best-estimate approach being considered for addressing NUREG-0737, Items II.K.3.30 and II.K.31
- Evaluations performed at the request of Maine Yankee for a reduced steam generator pressure
- Maine Yankee-specific SBLOCA application of RELAP5YA (YAEC-1868)

Upon completion of the review and interviews, the team discussed a number of questions drafted by the law firm of Winston & Strawn to ensure that specific concerns identified in the NRC's Demand For Information (DFI) letter had been addressed. Overall observations and conclusions were then formulated by the team relating to the development of the RELAP5YA model and subsequent work on and application of the evaluation model to Maine Yankee.

At the conclusion of the review, an exit debriefing was conducted by the Review Team Leader to present preliminary results of the review. Refer to Section C of the report for list of attendees.

E. Background

In the NRC's DFI letter addressed to Duke Engineering & Services, Inc./Yankee Atomic Electric dated December 19, 1997, the NRC states that engineering services provided to Maine Yankee for SBLOCA analysis were not in compliance with regulatory requirements. The NRC also states that Yankee Atomic, acting as a vendor for Maine Yankee, performed the inadequate analyses. This technical review was chartered to provide an independent review of the adequacy and compliance with 10CFR50.46 and 10CFR50 Appendix K of the subject evaluation model at the time of development and application to Maine Yankee, evaluate the cognizance of Yankee Atomic personnel of the model limitations, determine the appropriateness and timeliness of corrective actions that may have been taken, and to determine whether or not the model limitations were effectively communicated to Maine Yankee and the NRC.

The Review Team worked in conjunction with two attorneys from Winston & Strawn in providing input for DE&S's response to paragraph IV.A of the NRC's DFI. The Review Team also assisted the Winston & Strawn attorneys in their investigation in support of their response to paragraph IV.B of the DFI.

III. REPORT DETAILS

A. Yankee Atomic SBLOCA Methods Development History

Yankee Atomic initiated the RELAP5/MOD1, now called RELAP5YA, SBLOCA methods development project in 1980 to broaden its analytical capabilities to support their utility customers and Yankee Atomic's Rowe plant. Yankee Atomic was already performing non-LOCA analyses for Yankee Rowe, Vermont Yankee, and Maine Yankee. LBLOCA analyses based on the WREM package were also being performed as a result of a collaborative effort with Exxon Nuclear (now Siemens) as part of the transition to Exxon fuel supply for Maine Yankee. The RELAP5 program was a natural extension of the existing Yankee Atomic analysis capability, and was an attempt to upgrade to what Yankee Atomic considered the state-of-the-art SBLOCA code.

An additional consideration for developing these analytical capabilities in-house, rather than to continue to purchase analyses from vendors, was the uniqueness of the plant designs. Rowe in particular was a one-of-a-kind plant, and consequently there existed unique analysis requirements. Obtaining analyses from vendors over the life of the plant was also recognized as a significant economic consideration.

Yankee Atomic's decisions involving the development of in-house analysis capability were similar to those of many nuclear utilities. However, undertaking Appendix K LOCA methods was a very aggressive and challenging undertaking for any utility or engineering service organization. The decision by a utility to take SBLOCA evaluation model development responsibilities in-house has only been successfully completed by one other utility in the industry.

B. Maine Yankee SBLOCA Licensing Basis History

Original SBLOCA Licensing Basis

The original Final Safety Analysis Report (FSAR) Chapter 14 SBLOCA licensing basis calculations for Maine Yankee at 2440 MWt were performed by Combustion Engineering (CE). The LOCA break spectrum analyzed included large breaks, which were defined as the largest double-ended break down to 1.0 ft^2, and a different method for intermediate and small breaks. The SBLOCAs (intermediate and small breaks) were simulated with the BARF-II code for a break spectrum which included 0.55, 0.2, 0.1, 0.049, and 0.0218 ft^2 break sizes. The BARF-II code represented the Reactor Coolant System (RCS) primary with a two node model. For break sizes below approximately 0.07 ft^2 no core uncovery was predicted. For the larger break sizes the specific peak cladding temperatures (PCTs) were not given, but were summarized as not exceeding 2000°F.

Revised SBLOCA Licensing Basis (1971)

The original SBLOCA analysis was revised in 1971 with analyses performed by CE with the CEFLASH-4, PERC, and STRIKIN codes for the licensed power level of 2440 MWt. The methodology uses a multi-node representation of the RCS. In the document describing these re-analyses, the small break spectrum is described as "breaks with an area less than 0.5 ft^2," and the break spectrum analyzed consisted of 0.5, 0.2 and 0.1 ft^2 breaks. The PCTs were 1702°F for the 0.5 ft^2 break and 985°F for the 0.2 ft^2 break. The core did not uncover for the 0.1 ft^2 break size. For the break sizes which result in core uncovery the accumulator injection refloods the core and limits the PCT to acceptable values. Along with the LBLOCA analyses, these results met the requirements of the AEC Interim Acceptance Criteria for Emergency Core Cooling Systems.

Revised SBLOCA Licensing Basis (1977)

A revised SBLOCA spectrum analysis was performed in 1977 by CE using the 1974 CEFLASH-4AS model at a power level of 2630 MWt and a peak linear heat rate of 16.5 kW/ft. The purpose of these reanalyses was to establish a break spectrum which was consistent with the technical specification kW/ft limits. Prior analyses had used a lower kW/ft value for SBLOCAs. Break sizes of 0.5, 0.3, and 0.1 ft^2 were analyzed. The PCT results were 1348°F (0.5 ft^2), 1222 (0.3ft^2), and 1176 (0.1 ft^2). The 0.5 ft^2 case is mitigated mainly by the injection from the accumulators. The 0.3 ft^2 case is mitigated mainly by the HPSI pump, with the termination of the analyses justified by accumulator injection occurring. The 0.1 ft^2 case is mitigated solely by HPSI pump injection. These analyses were implemented with the Cycle 4 reload.

Maine Yankee SBLOCA Licensing Analyses Performed By Yankee Atomic Electric Company

Yankee Atomic submitted YAEC-1300P to the NRC for review in January 1983. This topical report describes the models and correlations in the RELAP5YA code used for LOCA analysis per 10CFR50 Appendix K. One of the objectives of this submittal was to respond to NUREG 0737, Item II.K.3.30 for Maine Yankee. While the NRC review of YAEC-1300P was in progress, the RELAP5YA code was applied by Yankee Atomic in an April 1984 submittal to the NRC to resolve the SBLOCA Reactor Coolant Pump (RCP) trip issue for Maine Yankee. The NRC issued an SER on the RCP trip study in April 1986. The SER on the YAEC-1300P topical report was issued on January 30, 1989. This SER included twelve conditions, among which was a requirement to submit future plant-specific licensing applications. By letter dated May 8, 1989, the NRC recognized that its SER closed out II.K.3.30. Although no submittal of plant applications had been made, the same letter closes out II.K.3.31. The letter stated that implementation may be the subject of a future inspection by NRC.

Application of the RELAP5YA code as described in YAEC-1300P to Maine Yankee was completed in 1993. The results of these analyses are summarized in YAEC-1868 dated June 1993. A break spectrum of 0.10, 0.15, 0.20, 0.25, 0.30, and 0.35 ft^2 was

analyzed. The 0.35 ft^2 case was not able to be run through to the final PCT. The final PCTs given in the report for the 0.35 ft^2 breaks are estimated based on a 58°F ΔPCT obtained from a 0.30 ft^2 break run, which was added to the calculated PCT at the time of safety injection tank actuation. The limiting PCT result was determined to be 1887°F for the 0.15 ft^2 break. PCTs for smaller and larger breaks were lower, thereby indicating a local PCT peak within the small break range. This report was not submitted to the NRC for review. Maine Yankee and Yankee Atomic have taken the position based on the May 8, 1989 NRC letter that submittal was no longer required, and that having the documents on file for NRC inspection was sufficient.

Implementation of the YAEC-1868 analysis results occurred with the Cycle 14 reload. The Core Performance Analysis Report for Cycle 14, which describes the RELAP5YA SBLOCA analysis results, was submitted to the NRC. This reload submittal replaced the 1977 Cycle 4 CE SBLOCA analyses with the YAEC analyses as the licensing basis analyses of record.

C. Maine Yankee SBLOCA Analysis Comparisons

Licensing Basis Analyses By CE

The SBLOCA analyses that established the licensing basis prior to Cycle 14 were performed by CE. The original licensing basis analyses were performed with a two node model and are not considered to be relevant. The revised SBLOCA spectrum analysis performed in 1977 used the 1974 CEFLASH-4AS evaluation model at a power level of 2630 MWt and a peak linear heat rate of 16.5 kW/ft. Break sizes of 0.5, 0.3, and 0.1 ft^2 were analyzed. The PCT results were 1348°F (0.5 ft^2), 1222°F (0.3 ft^2), and 1176°F (0.1 ft^2). These analyses were implemented with the Cycle 4 reload. The PCT vs. break size trend increases with break size and does not indicate a local PCT peak within the small break range.

1996 Analyses By Siemens Power Corporation

Siemens was contracted by Maine Yankee in 1996 to supply a SBLOCA analysis with the intent of lifting the NRC restriction to 90% full power operation. Siemens uses an NRC-approved Appendix K evaluation model that is based on RELAP5/MOD2. Siemens ran break sizes of 0.05, 0.10, 0.15, 0.20, 0.25, and 0.61 ft^2. The resulting PCTs were 1704°F (0.10 ft^2), 1522°F (0.15 ft^2), 1363°F (0.20 ft^2), 1157°F (0.25 ft^2), and 1382°F (0.61 ft^2). The core did not uncover for the 0.05 ft^2 case. Compared to the Yankee Atomic results in YAEC-1868, the Siemens results show a shift in the worst break size from 0.15 to 0.10, and a decrease in the limiting PCT from 1887°F to 1704°F. The Siemens results also show one local PCT peak, whereas the Yankee Atomic results show a second small local PCT peak. No comparison can be made at the 0.61 ft^2 break size since the Yankee Atomic model was unable to analyze that case. It is

noted that the Siemens prediction for 0.61 ft^2 does indicate an increase in the PCT relative to the 0.25 ft^2 break.

1996 Analyses By ABB

ABB was contracted by Maine Yankee in 1996 to supply SBLOCA analyses for Cycle 15 at a power level of 2440 MWt and 14.1 kW/ft. ABB used their 1977 Supplement 1 NRC-approved SBLOCA evaluation model. The break spectrum analyzed included 0.10, 0.15, 0.20, and 0.50 ft^2 break sizes. This range of break sizes is consistent with the ABB ECCS performance evaluation methodology. The PCT results are 1134.7°F (0.10 ft^2), 1405.0°F (0.15 ft^2), 1290.9°F (0.20 ft^2), and 1520.7°F (0.50 ft^2). The limiting break size remains the same as the 1977 CE analysis, but a local peak at 0.15 ft^2 is new to this reanalysis. It is noted that many of the assumptions in the reanalysis have changed and contribute to the differences with the 1977 analysis. The latest ABB methods do not agree with either the Yankee Atomic or the Siemens analyses in terms of the limiting break size.

WREM-Based Analysis By Yankee Atomic (1977)

In support of Cycle 5 Yankee Atomic performed an analysis of the 0.5 ft^2 break with the NRC-approved WREM-based LBLOCA evaluation model. This analysis was submitted to the NRC with the Cycle 5 technical specification change package, which was approved by the NRC. The analysis was performed to show that the transition to Exxon (Siemens) fuel did not adversely affect the SBLOCA analysis of record. The 0.5 ft^2 break was the limiting break size for the CE licensing basis analysis of record. The WREM-based analysis is characterized as a "demonstration analysis" in the licensing documentation, and was not intended to replace the CE analysis. The result of the analysis at 16.0 kW/ft was a PCT of 1229.9°F, therefore indicating that the larger SBLOCA break sizes were less limiting.

RELAP5/MOD2 & MOD3 Analyses By Yankee Atomic

Yankee Atomic has analyzed SBLOCAs for Maine Yankee with the more advanced RELAP5/MOD2 and MOD3 codes. In 1990 the Yankee LOCA staff gained experience by running the Siemens Appendix K SBLOCA evaluation model version of RELAP5/MOD2. This experience provided modeling guidance related to the ECCS bypass phenomenon. The Siemens model did not calculate the excessive ECCS bypass as was occurring in the Yankee Atomic model. The RELAP5/MOD3 version used by Yankee Atomic does not include the Appendix K models and has not been submitted for NRC review, but the results are useful for comparison. These analyses were important mainly in that they demonstrated that the PCT occurred shortly after accumulator injection, and then steadily decreased. This information supported the Yankee Atomic interpretation that the unstable RELAP5YA behavior following accumulator injection

was non-physical, and that the unstable PCT behavior following accumulator injection was not significant.

Summary of Comparisons

All of the SBLOCA analyses confirm that the LBLOCA PCTs are limiting. The Yankee Atomic, Siemens, and 1996 ABB methods all predict a local peak in the range of 0.10 to 0.15 ft^2. The ABB analyses and the CE analyses with both the 1974 and 1977 evaluation models predict that the PCT is highest for the 0.50 ft^2 break. The Siemens model predicts that the PCT increases as the break size approaches 0.61 ft^2, but that it is non-limiting relative to the local PCT peak at 0.10 ft^2. This trend cannot be confirmed from the Yankee Atomic RELAP5YA analyses due to no breaks being analyzed from 0.35 to 0.50 ft^2 due to either code execution difficulties or no cases having been run. The Yankee Atomic position is that the 0.5 ft^2 break PCT is bounded by the 0.15 ft^2 PCT of 1887°F. Given the information provided and based on the trends of other analyses, the Review Team was unable to draw a definitive conclusion regarding the RELAP5YA PCTs for the unanalyzed portion of the Maine Yankee SBLOCA spectrum. However, based on all the available calculations, it is concluded that the SBLOCA PCTs for all of these analyses meet the 10CFR50.46 2200°F criterion, and that SBLOCAs remain bounded by LBLOCAs.

D. Review Team Observations and Conclusions

1. There was a failure of the Yankee Atomic organization as well as the LOCA Group in fully understanding NRC expectations (in particular the NRC's Reactor Systems Branch) for an Appendix K evaluation model, as understood within the industry LOCA community, for 10CFR50.46 model acceptance. Plant-specific calculations were undertaken too late to change the direction of the development project following the identification of significant code limitations.

The YAEC-1300P RELAP5YA SBLOCA code was approved by the NRC in 1989. However, the development and NRC approval process was not without risk in that no plant applications were performed in parallel with the development of the code and model. It is the standard industry practice that the plant applications be performed to identify any problems prior to submitting the code and evaluation model for NRC review. Limiting the testing and validation of the model to scaled test facilities does not challenge the method sufficiently, since some code and model problems will invariably arise with the first plant applications. Any applications should be completed utilizing a process that meets NRC's expectations prior to any implementation in the plant's licensing basis The Review Team's understanding of NRC's expectations is based on many years of LOCA-related licensing interactions. These expectations are evident in the Demand for Information Letter and in the

transcript of the December 18, 1995 meeting between the NRC and Yankee Atomic.

The NRC's intent regarding the conditions attached to the YAEC-1300P SER was to ensure that plant applications of RELAP5YA would be reviewed and approved prior to implementation. This is a standard process for approving LOCA methods applications. Variations on this process are possible, but are subject to the approval of the NRC. Yankee Atomic should have realized that this step in the approval process was the standard industry practice despite the NRC letter dated May 8, 1989.

Yankee Atomic failed to bring to the attention of the NRC several significant problems that were encountered in the application of the approved YAEC-1300P method to Maine Yankee. The code results were unstable following accumulator injection for much of the break spectrum. The model could not run through the historical break spectrum up to 0.5 ft^2, in particular when considering that the limiting PCT case for the current licensing basis was 0.5 ft^2. The NRC expects that the approved evaluation model should be capable of running any case should the need arise. The Review Team acknowledges and agrees with the NRC position. Partial break spectrums are intended to reduce the number of required LOCA calculations when agreement can be reached that all of the critical challenges to the ECCS system have been covered. The reduced spectrum is not intended to provide for inadequacies in the analysis methods. An exception could be made but only with NRC concurrence. Yankee Atomic asserts that its break spectrum was sufficient to meet the licensing requirements. The Review Team acknowledges, that partial break spectrums are licensable; however, the method must be capable of calculating all break sizes in the spectrum. Assuming rather than confirming that the Yankee Atomic break spectrum was acceptable, was a failure in meeting NRC expectations on the part of Yankee Atomic.

Excessive ECCS bypass (a code deficiency) was occurring in the SBLOCA analyses. To address the excessive ECCS bypass, Yankee Atomic used a large non-physical loss coefficient for the junction between the two reactor vessel downcomer volumes. Yankee Atomic considers this large non-physical loss coefficient to be an input to the code rather than a model change. The Review Team disagrees with Yankee Atomic's rationale. The change comprises a manipulation of the intended use of the input parameter in that it is altered sufficiently to nullify the original intent of the parameter as described in the code documentation. The Review Team considers this to be a modeling change that should have been discussed with the NRC before application.

Application of the unapproved best-estimate RELAP5YA methodology to the steam generator pressure reduction issue without NRC concurrence was improper. The Yankee Atomic LOCA Group staff applied the best estimate methodology

since it was the only available in-house analysis tool. This was done without a LOCA Group staff or Yankee Atomic management awareness or understanding of the NRC approval status of the best-estimate model, the Maine Yankee licensing basis, or NRC expectations.

The above shortcomings in Yankee Atomic's program for performing and implementing the Maine Yankee SBLOCA application are all related to an unfamiliarity with the NRC's expectations for the LOCA analysis approval process as understood by the industry LOCA community. Although these expectations are not completely documented in specific references, such as the NRC Standard Review Plan or Regulatory Guides, they have been clearly communicated to the industry's LOCA community through extensive and numerous interactions with the NRC (the Reactor Systems Branch, the Advisory Committee on Reactor Safeguards (ACRS), and others) since the Interim Acceptance Criteria of 1971. These interactions have made it clear to what degree the community was to keep the NRC informed, what types of issues needed NRC approval, and what the NRC expectations were for approved evaluation models. Yankee Atomic would have benefited from the experience of companies or individuals who more fully participated in the process. The Review Team recognizes that it remains Yankee Atomic's responsibility to understand NRC's expectations. Yankee Atomic would also have benefited from improved, continuous and direct technical communications with the NRC Reactor Systems Branch, which could have eliminated many of the issues of concern as identified by the Review Team. This situation could have been avoided with sustained communications and developing a continual working relationship with the NRC technical staff, or by consulting with industry LOCA experts. The Review Team does note that Yankee Atomic has apparently been successful in licensing LOCA analysis methods for Yankee Rowe and Vermont Yankee.

2. ***During the development of the Maine Yankee specific SBLOCA model and analyses there were a number of organizational barriers and a culture which impeded the LOCA Group from completing the project in a manner which met NRC expectations.***

Root causes for the problems that occurred during the development of the Maine Yankee specific SBLOCA model and analyses are associated with organizational barriers and the culture in the Yankee Atomic organization. Although Yankee Atomic upper management was aware of the RELAP5YA SBLOCA development project and technical and licensing issues, there was insufficient upper management involvement and direction. Due to its high visibility and legal status in the NRC regulations, a LOCA development program requires a very significant resource commitment which includes a sustained critical mass of expertise and continuity of key personnel. In the Review Team's opinion, Yankee Atomic management did not obtain sufficient experience and depth in the staffing of the RELAP5YA project

team, in particular at the beginning of the project in the early 1980's. Few of the staff had any prior LOCA analysis experience. This comment is based on the knowledge of the resources that other LOCA analysis organizations employ on similar projects. The Review Team notes that Intermountain Technologies Inc. was contracted with to install many of the Appendix K models into RELAP5YA. The RELAP5YA project team was working in relative isolation from the LOCA community with a code that only one other organization was applying for licensing-basis SBLOCA analysis. The culture at Yankee Atomic, which appears to prefer to be self-sufficient and independent in providing engineering services, contributed to the isolation of the LOCA Group and the failure to appreciate the LOCA licensing implications. No benchmarking to other industry LOCA organizations occurred. A final contributing factor was computer resource limitations. Prior to approximately 1992 the LOCA engineers were unable to run all of the desired cases to fully develop and test the code and models, and the execution time was too slow for production work.

With the benefit of hindsight, the original RELAP5/MOD1 code limitations presented too great a technical challenge to the Yankee LOCA group in achieving an Appendix K model that could be used with confidence. The project was further challenged by the resource limitations, ineffective management, and the independent culture at Yankee Atomic. These problems appear to be limited to the RELAP5YA SBLOCA application to Maine Yankee, as these problems did not surface on the RELAP5YA licensing for Rowe and Vermont Yankee. This comment is based on the Review Team's understanding of the thorough inspections that have occurred following the allegations in December 1995, and the absence of any other significant problems.

3. ***The Yankee Atomic organization was aware of performance problems with the RELAP5YA code. These code problems were addressed by the LOCA Group from a technical perspective, however the organization did not adequately maintain communications with the NRC technical staff. This resulted in a failure to resolve all technical issues necessary to satisfy NRC expectations prior to implementation for the Cycle 14 reload.***

The RELAP5/MOD1 code, which was the starting point for the RELAP5YA version, has inherent limitations which complicate the application for an Appendix K SBLOCA evaluation model suitable for NRC approval. Long-standing code deficiencies were identified during the project. The Review Team considers completely addressing these code limitations to be a significant challenge to the Yankee Atomic LOCA Group. RELAP5YA's 5-equation formulation does not appropriately represent non-equilibrium behavior. This formulation lags the state-of-the-art and can result in excessive condensation and vaporization. To compensate for this deficiency an artificially high (200°F) ECCS water temperature was introduced. Excessive prediction of interfacial drag caused non-physical ECCS

bypass that severely hindered SBLOCA modeling. The drag model was improved but the problems persisted for the larger break sizes. In a further attempt to address the ECCS bypass problem a very large non-physical loss coefficient was introduced between the two volumes representing a split reactor vessel downcomer. This approach was supported by Yankee Atomic by verification to experiments with relevant data. The Review Team considers this approach to be an acceptable compensation for a RELAP5YA code deficiency, and the use of a value of 600 obtained an amount of ECCS penetration that was consistent with industry experience. The preferred approach would have been to implement the ECCS penetration modeling as a code change. The Review Team also considers this modeling approach, whether termed a model change or an input change, to be significant and should have been discussed with the NRC.

Yankee Atomic also experienced large changes in RELAP5YA results with only small changes in inputs. The code was excessively sensitive and PCTs were inconsistent. The code and model development project was concluded without completely addressing the underlying code limitations. It was decided at least annually to continue with the licensing of RELAP5YA given these code and modeling limitations, and then to implement the results documented in YAEC-1868 without NRC review. The Review Team questions the advisability of these decisions given the expectations of the NRC in the LOCA arena. It is noted that there was no real inappropriate use of YAEC-1300P prior to the implementation of the methodology for Cycle 14. Continuation of the communications with the NRC technical staff that were established and functional during the review of YAEC-1300P could have been a path towards acceptable resolution of these issues.

Since implementation for Cycle 14 proceeded without the capability to analyze the full SBLOCA break spectrum, NRC expectations as understood by the industry LOCA community were not met. Yankee Atomic's position that their analyzed break spectrum was complete in that it bounded the limiting SBLOCA PCT is understandable, but is not consistent with standard industry practice since all break sizes could not be calculated and did not address the limiting break size for the analysis of record.

4. ***The technical capabilities of the Yankee Atomic LOCA Group are acceptable for performing LOCA analysis. The LOCA Group was qualified and knowledgeable regarding SBLOCA computer models and phenomena. Interviews and documents did not indicate any evidence of deliberate violations of NRC regulations.***

Based on interviews and documentation reviews, the Review Team concludes that the Yankee Atomic LOCA Group is knowledgeable about SBLOCA computer models and phenomena, and that the staff are technically qualified. The project included extensive assessment of RELAP5YA to scaled test facility SBLOCA data.

The LOCA Group staff interviewed are sincere, honest, open, and cooperative. Throughout the history of the RELAP5YA project the staff has been conscientious and hardworking, and had good intentions of establishing independent analysis capabilities to broaden their support of their customers. It was also apparent to the Review Team that the LOCA Group staff was not expert in LOCA analysis-related history, precedent, and licensing processes. No evidence of deliberate violations of NRC regulations were identified.

5. ***Based on the information available, including other analysis results, it can be concluded that the results of Yankee Atomic's SBLOCA calculations for Maine Yankee are consistent with the results of some other calculations by other codes and organizations. Therefore it is concluded that the SBLOCA PCTs for Maine Yankee meet the 2200 °F criterion of 10CFR50.46, and that SBLOCAs remain bounded by LBLOCAs for this plant.***

The results of the Maine Yankee RELAP5YA analyses documented in YAEC-1868 include instabilities related to the limitations of the code. These instabilities mainly occur following accumulator injection, which occurs after a significant uncovery and heatup of the cladding. Yankee Atomic has concluded that the predicted PCTs are conservative, and that the instabilities do not affect the prediction of the PCT for the limiting break size (1887°F for the 0.15 ft^2 break). Yankee Atomic has also concluded that the inability to calculate the larger break sizes up to 0.50 ft^2 is not significant since the larger break sizes are non-limiting. The results of other calculations have been evaluated by the Review Team. The prior SBLOCA analyses performed by CE in 1977 with the 1974 SBLOCA evaluation model have lower PCTs, with an increasing PCT with break size trend (1348°F for 0.50 ft^2 break). The 1996 comparison calculations performed by ABB with the 1977 SBLOCA evaluation model predict a PCT of 1520.7°F for the 0.50 ft^2 break. The ABB analysis also showed a local PCT peak of 1405°F at 0.15 ft^2. It is noted that the analysis inputs for the 1977 CE and 1996 ABB analysis do not match the Yankee Atomic or the Siemens analyses, but are still useful for comparison purposes. The 1996 comparison calculations performed by Siemens show a PCT of 1704°F for the 0.10 ft^2 break, and a PCT of 1382°F and an increasing PCT trend at 0.61 ft^2. Other Maine Yankee SBLOCA analyses were performed by Yankee Atomic in 1977 using the WREM-based model, in 1990 using the Siemens RELAP5/MOD2 model, and in 1995 using RELAP5/MOD3. The WREM-based demonstration calculation indicated a low PCT relative to LBLOCAs. Experience gained with the Siemens model helped address the ECCS bypass problem. The RELAP5/MOD3 best estimate analyses, although not NRC approved, predicted a SBLOCA transient response which is similar to RELAP5YA following accumulator injection. The comparison of all of the SBLOCA results for Maine Yankee, including the Yankee Atomic RELAP5YA results, indicates that the PCTs for SBLOCA are less than 2200°F, and that SBLOCAs are bounded by LBLOCAs. Therefore there was no reduction in the safety of the plant.

6. ***Yankee Atomic's direct interface with the NRC on the application of the RELAP5YA model appears to have ended once the SER was received in 1989. Not maintaining direct a interface between the LOCA Group and the NRC technical staff resulted in a loss of communications regarding NRC's expectations for SBLOCA calculations performed with RELAP5YA to support the Maine Yankee license.***

Communications with the NRC during the NRC review of the YAE-1300P topical report appear to be adequate, since NRC approval was eventually obtained. The Review Team notes that the large number of conditions in the SER and the substantial nature of the conditions are not positive or desirable results following a multi-year NRC review, and indicates, in part, code weaknesses. Yankee Atomic then developed responses to the conditions and documented them internally. The Maine Yankee applications completed in 1993 and documented in YAEC-1868 were not submitted for NRC review. This decision was based on the May 8, 1989 NRC letter which closed out NUREG-0737 Item II.K.3.31. Although this NRC letter conflicted with the YAEC-1300P SER conditions requiring plant applications to be submitted, no Yankee Atomic communication with the NRC occurred. The Review Team views this as a major breakdown in the licensing function, and an absence of awareness of NRC expectations. The major issues that have resulted evolved, in part, from this lack of communication and coordination with the NRC.

The root cause of this lack of communication and coordination with the NRC is the apparent transfer of responsibility for NRC interface on LOCA matters from Yankee Atomic to Maine Yankee following the issuance of the YAEC-1300P SER in 1989. Due to the high visibility and regulatory significance of LOCA analysis, it is essential to maintain an interface between the LOCA analysis group and the NRC technical staff. This did not occur. At the major vendor LOCA analysis organizations this NRC interface is given a high priority and is maintained because the analytical model is the responsibility of the vendor. In discussions with the Yankee Atomic Project Manager for Maine Yankee it was stated that Maine Yankee directed Yankee Atomic not to directly interface with the NRC. It has become the standard industry practice for a licensee to insist on vendor interactions with the NRC, associated with the licensee's docket, to be through the licensee. However, this constraint, as understood by Yankee Atomic, contributed to the licensing process breakdown in this highly technical subject area.

7. ***Continuous communications with the NRC could have eliminated many of the issues of concern.***

Subsequent to initial NRC approvals, issues have arisen and errors have been discovered in industry LOCA evaluation models and plant applications. As issues or errors are identified within the industry or by a LOCA analysis organization, it is the

recognized responsibility of each organization to respond and communicate with the NRC. These communications are essential for problem identification, initiating appropriate corrective actions, and maintaining good technical credibility with the NRC. Due to the complex technical nature of LOCA analysis, it is necessary that these issues be communicated and discussed by the experts. It is necessary that the organization appreciate and recognize the significance of staying within the LOCA regulations and in understanding the NRC's expectations in that regard. Historically, good communications with the NRC has enabled reasonable and fair regulation as LOCA issues are dispositioned. The Review Team concludes that successful communications between the Yankee Atomic LOCA Group and the NRC technical staff could have eliminated many of the issues of concern related to Maine Yankee SBLOCA analyses.

IV. TECHNICAL REVIEW TEAM'S CONCLUSIONS ON ISSUES IDENTIFIED IN NRC'S DEMAND FOR INFORMATION

A. Summary of Apparent Violation as Stated in Demand for Information

During Cycle 14 operations, and in the Cycle 14 and Cycle 15 CPAR analyses, YAEC caused Maine Yankee to use apparently unacceptable evaluation models which could not calculate or reliably calculate ECCS performance. The models used were in apparent violation of 10 C.F.R. § 50.46(a)(1), because there was a region of the small-break spectrum between break sizes of 0.35 ft^2 and at least 0.6 ft^2 for which no acceptable evaluation model could either calculate or reliably calculate ECCS performance. The Manager and the Lead Engineer knew of this. In addition the oscillations and instability in the analysis became more severe at larger break sizes, increasing the risk that the limiting breaks had not been identified.

Review Team's Conclusion on Item A

The Maine Yankee SBLOCA analysis performed by Yankee Atomic with the RELAP5YA code consisted of a spectrum of break sizes ranging from 0.1 ft^2 to 0.35 ft^2. Yankee Atomic was not able to run the 0.35 ft^2 break size through to the final PCT. The final PCTs given in YAEC-1868 for the 0.35 ft^2 breaks are estimated based on a 58°F ΔPCT obtained from a 0.30 ft^2 break run, which is added to the PCT calculated at the time of safety injection tank actuation. Yankee Atomic concluded that the limiting SBLOCA PCT was determined to be within this range of break sizes, and determined a limiting PCT value of 1887°F for the 0.15 ft^2 break. The rationale for this conclusion was documented as being based on the decreasing trend of PCT for smaller and larger break sizes. Yankee Atomic maintained that this scope of SBLOCA analyses met the 10CFR50 Appendix K and 10CFR50.46 regulations. This interpretation was consistent with Yankee Atomic's understanding of the regulations. In response to questions from the Review Team, the LOCA staff provided additional verbal justification consisting of an explanation of SBLOCA phenomena which supported their conclusion. Additional

analysis results from other codes have been presented by Yankee Atomic which tend to support the Yankee Atomic position. It is noted by the Review Team that some of the trends in the additional analyses are not consistent with the Yankee Atomic position.

The prior licensing basis SBLOCA analysis performed by Combustion Engineering in 1977 established a trend of increasing PCT with break size, with the 0.5 ft^2 break having a PCT of 1348°F. Other non-licensing basis analyses performed later by Siemens and ABB indicate both a local PCT peak near the limiting Yankee Atomic break size, and a trend of increasing PCT at the largest break size analyzed. Given the information provided and based on the trends of other analyses, the Review Team was unable to draw a definitive conclusion regarding the RELAP5YA PCTs for the unanalyzed portion of the Maine Yankee SBLOCA spectrum. However, based on all of the calculations, it is concluded that the SBLOCA PCTs for all of these analyses meet the 10CFR50.46 2200°F criterion, and that SBLOCAs remain bounded by LBLOCAs.

The Review Team agrees that the industry standard practice is consistent with the NRC's position that an Appendix K SBLOCA evaluation model must be capable of analyzing any break size within the plant's SBLOCA licensing basis. This does not mean that all break sizes need to be analyzed, but rather that the model must be capable of analyzing them. The RELAP5YA evaluation model has not demonstrated the capability to analyze the historical Maine Yankee SBLOCA break spectrum. To meet the expectations of the NRC, sound engineering arguments can be used but should be communicated to the NRC and agreed upon prior to implementation.

The Review Team agrees with the NRC's position that the Yankee Atomic SBLOCA model produced oscillatory and unstable results. This behavior is evident following accumulator injection and in particular for the larger break sizes. These code problems were long-standing and widely known within the Yankee Atomic organization. The model is also considered by the Review Team to be unreliable in that an unexpected large change in results can occur with only a small change in the input to the code. Yankee Atomic considered the oscillations and instability to be non-physical, and that the PCTs predicted were sufficient. The Review Team believes that the oscillations and instability may, in part, be non-physical and due to fundamental limitations in the RELAP5YA code. Based on a broader knowledge of SBLOCA phenomena and results from other codes, Yankee Atomic was confident that SBLOCA was bounded by LBLOCA. Based on this expectation Yankee Atomic accepted the results from RELAP5YA as adequate for showing compliance with the regulations. The Review Team understands that this may have been a correct conclusion. However, Yankee Atomic's conclusion is based, in part, on information beyond the demonstrated results of runs of the RELAP5YA code for Maine Yankee. The Review Team concludes that this situation should have been communicated to the NRC prior to implementation.

The Yankee Atomic report YAEC-1868, which summarizes the results of the application of the SBLOCA evaluation model to Maine Yankee, discloses the results of

the analyses, including the break spectrum analyzed, that the 0.35 ft^2 break case would not run through, and Yankee Atomic's explanation as to why the results meet the regulations. The illustrated results show the unstable and oscillatory behavior of the code. This report was reviewed by Yankee Atomic management and approved. The Review Team understands that YAEC-1868 was not submitted for NRC review based on a communication from the NRC Project Manager for Maine Yankee and an expectation that NRC review would be by a future inspection. Not submitting YAEC-1868 for NRC review was an error in that it should have been recognized that NRC review was necessary prior to implementation.

As a consequence of the breakdown in the LOCA licensing process which resulted in implementation of the SBLOCA analyses for Maine Yankee Cycle 14 prior to obtaining NRC review and approval, Yankee Atomic did provide Core Performance Analysis Reports (CPARs) to Maine Yankee with the deficiencies described above. This situation was not a result of any deliberate action to avoid compliance with the regulations, but rather failure of the Yankee Atomic organization to understand NRC expectations. The Review Team does not consider this organizational failure to have resulted in a reduction in the safety of the plant, since LBLOCA is limiting and sets the core operating limits. Had the SBLOCA analyses been completed it is likely that the core operating limits for Cycle 14 would have remained the same.

B. **Summary of Apparent Violation as Stated in Demand for Information**

As a result of YAEC's preparation and review of YAEC-1868, YAEC provided MYAPCo with information that was not complete and accurate in all material respects, and thus caused MYAPCO to be in apparent violation 10 C.F.R. § 50.9(a). CPARs maintained for information and submitted to the NRC by MYAPCo, in support of Cycle 14 and Cycle 15 reload applications were apparently not complete and accurate in all material respects. MY relied on YAEC-1868 to prepare Cycle 14 and Cycle 15 CPARs in order to demonstrate compliance with 10 C.F.R. § 50.46. YAEC-1868, in its entirety, conceals the lack of an acceptable evaluation model to calculate ECCS performance for a portion of the break spectrum between 0.35 ft^2 and at least 0.6 ft^2. As a result of the OI investigation it appears that no Maine Yankee personnel realized that the RELAP5YA code failed at 0.35 ft^2 or that there might be a portion of the break spectrum for which there was no acceptable evaluation model to calculate ECCS performance, and that no one at YAEC informed MYAPCo personnel that RELAP5YA had failed at 0.35 ft^2.

Review Team's Conclusion on Item B

YAEC-1868 documents the results of the SBLOCA analyses performed by Yankee Atomic using the NRC-approved YAEC-1300P SBLOCA evaluation model. The Review Team concludes that this document was sufficiently complete and accurate as a summary of the SBLOCA calculations that were performed. However, the downcomer

modeling changes that were not discussed in YAEC-1868 should have been communicated to the NRC in YAEC-1868, as a revision to YAEC-1300P, or in some other communication. It is clear that Yankee Atomic did not appreciate the regulatory significance of the downcomer modeling changes. YAEC-1868 was understandable to its intended audience, and was suitable for a licensing submittal in support of Maine Yankee. It is noted that the abstract is potentially misleading in the use of the word "complete", but that the scope of the analysis as contained within the report is characterized correctly. The amount of technical information included was appropriate for any knowledgeable person to understand the results of the analysis. There was no intentional concealment of information which would have identified any non-compliance with NRC regulations. The statements in the report regarding compliance with the NRC's regulations were consistent with the Yankee Atomic LOCA Group's understanding of the regulations. The Review Team concludes that Yankee Atomic's understanding of the regulations was not consistent with the understanding of peers in the industry LOCA community. The compliance statements and the supporting analyses in YAEC-1868, as interpreted by a knowledgeable engineer not trained in the LOCA licensing process, could be understood as a logical basis for compliance. Therefore, YAEC-1868 could be understood by a knowledgeable engineer not trained in the LOCA licensing processes to be complete and in compliance with the regulations. It is likely that the compliance statements and the supporting analyses in YAEC-1868 would be understood by an NRC reviewer and would have led to interactions with Yankee Atomic, in particular since the limiting break size in the analysis of record was 0.5 ft^2 and was not analyzed with RELAP5YA.

Based on discussions with Yankee Atomic personnel, the problems with applying the RELAP5YA code to Maine Yankee were a topic of discussion over several years with cognizant Maine Yankee personnel. Although utility staff are generally not LOCA experts, people assigned to interface with LOCA organizations generally have sufficient knowledge to understand the subject, to review associated documents, and to make appropriate decisions on behalf of the utility. However, utility staff generally rely on the LOCA organization to manage the technical details associated with compliance with Appendix K and 10CFR50.46. The Review Team concludes that Maine Yankee staff, similar to typical utility staff, should have relied on the Yankee Atomic LOCA organization for LOCA licensing support. Yankee Atomic understands that Maine Yankee directed it not to independently interface with the NRC staff. As a result, there was no Yankee Atomic interface with the NRC on LOCA issues. The Review Team maintains that the code and its models are Yankee Atomic's responsibility. Had Yankee Atomic management been more aggressive and achieved the continuation of communication with the NRC, Yankee Atomic personnel would have been better prepared to make decisions relative to NRC expectations with the likely result that communications on the deficiencies of YAEC-1868 would have occurred.

C. Summary of Apparent Violation as Stated in Demand for Information

During Cycle 14 operations and in the Cycle 14 and Cycle 15 CPAR, YAEC caused MYAPCo to use an apparently unacceptable SBLOCA evaluation model which over-predicted core cooling. YAEC-1868 apparently did not satisfy the requirements of 10 C.F.R. § 50.46(a)(1) because as a result of incorrect calculations of the penetration factors, which arose from misapplication of the Alb-Chambre penetration correlation, the analysis provides no basis to assume full penetration of the emergency core cooling system injection and provides no basis to derive the loss coefficient of 600 used for the split downcomer nodalization. These deficiencies resulted in over-prediction of core cooling and overstatement of the conservatism of the model. If the Alb-Chambre correlation had been applied correctly, penetration factors would have been calculated in the range of -0.6657 to -0.7767, which is a meaningless result because the calculations would have been less than zero. Such calculations also indicate other possible errors in application of the Alb-Chambre correlation. Adequate QA review would have revealed the errors and the unacceptability of the RELAP5YA SBLOCA analysis described in YAEC-1868.

Review Team's Conclusion on Item C

Early applications of the RELAP5YA SBLOCA evaluation model to Maine Yankee identified excessive ECCS bypass relative to what was expected based on scaled test facility data and the results of other codes. Revisions to the interfacial drag model were only partially successful in addressing this non-physical prediction. Various modeling approaches were attempted to make the ECCS penetration into the vessel lower plenum more physical. Eventually an artificially large loss coefficient was introduced in the junction connecting the two volumes representing a split reactor vessel lower downcomer. The value of this loss coefficient was varied to obtain a balance between the expected ECCS penetration and the effect on steam venting via the break. A value of 600 was selected as an appropriate value. The amount of ECCS penetration obtained with this modeling approach was justified, in part, on the Alb-Chambre correlation. This correlation was applied to confirm that the amount of ECCS penetration predicted by RELAP5YA was conservative. The Review Team considers this approach to be an acceptable compensation for a RELAP5YA code deficiency, and the use of a value of 600 obtained an amount of ECCS penetration that was consistent with industry experience. The preferred approach would have been to implement the ECCS penetration modeling as a code change. This modeling approach is not expected to result in an overprediction of core cooling, but since the calculations were not completed for all break sizes, it cannot be definitively confirmed.

An error was made in the application of the Alb-Chambre correlation. This arithmetic error was not identified during the quality assurance process. This is a failure of the quality assurance process. Closer investigation of this failure offers a reasonable explanation as to why it occurred. The arithmetic error did not skew the result of the

calculation, which was in the range of the expected result that complete penetration was predicted. The correlation can produce results, in excess of the value of 1 (in this case a value of 8), which have the meaning of complete ECCS penetration. A person performing a quality assurance review is influenced by the result based on experience and expectations. These are the most difficult errors to identify. A correct application of the Alb-Chambre correlation without the arithmetic error would have produced negative values indicating complete ECCS bypass. This result would have been immediately recognized by Yankee Atomic as non-physical for the SBLOCA conditions of interest. The cause of the non-physical result would have been traced to excessively conservative input values. More reasonable values would then be input to the correlation, and reasonable and valid results indicating significant ECCS penetration would have resulted. Therefore, although the application of the Alb-Chambre correlation was in error, the basis for incorporating the loss coefficient with a value of 600 remains valid. Thus the results of the error in applying the Alb-Chambre correlation did not result in invalid input to the SBLOCA analyses.

The Review Team concludes that the use of the Alb-Chambre correlation as a confirmation of the modeling approach which includes the junction loss coefficient of 600 in the reactor vessel downcomer is reasonable given the available data and the deficiency of the code. The Review Team recognizes that the PCT results of the SBLOCA analyses for some of the break sizes are very sensitive to the value of the loss coefficient. This entire modeling approach was not presented to the NRC for review. Yankee Atomic considers the value of the downcomer junction loss coefficient to be an input to the evaluation model. This is literally true, since all junction loss coefficients and most of the plant applications model are in the form of inputs. However, the Review Team concludes that due to the non-physical value used and the significance of this input parameter, it is in reality an important model change requiring NRC approval. The Review Team believes that this model could have been approved by the NRC in this form or with some revision.

D. Summary of Apparent Violation as Stated in Demand for Information

YAEC caused Maine Yankee to apparently violate 10 C.F.R. § 50.46(a)(1) by relying on an unacceptable SBLOCA evaluation model (Best Estimate RELAP5YA SBLOCA evaluation model) to calculate ECCS cooling performance in preparing a Section 50.59 analysis used to determine if a decrease in steam generator pressure involved an unreviewed safety question. Additionally, the proposed BE RELAP5YA evaluation model was not the approved SER computer code and did not comply with 10 C.F.R. Part 50, Appendix K requirements. The NRC indicates that is also reasonable to conclude that the Manager knew that the analysis which YAEC performed regarding the effects of a reduction in steam generator pressure on LOCA analyses was a safety analysis which would be used by Maine Yankee in a Section 50.59 analysis or other safety analysis.

In view of the intended use of the YAEC analysis, the Manager should have provided Maine Yankee with an analysis which met NRC requirements.

Review Team's Conclusion on Item D

Due to difficulties in applying the YAEC-1300P Appendix K RELAP5YA SBLOCA code to the Maine Yankee plant, the Yankee Atomic LOCA Group initiated a parallel effort to develop a "best-estimate" (BE) LOCA evaluation model, and indicated this in a memorandum. The BE modeling approach is an alternative approach to the traditional Appendix K approach. Since the YAEC-1300P topical report did not include this BE approach, separate NRC approval of this method would be necessary prior to implementation. The Yankee LOCA Group Manager clearly understood that the BE approach required further NRC review. The possibility of using the proposed BE approach to satisfy NUREG-0737 Item II.K.3.31 for Maine Yankee had been discussed with the Maine Yankee staff. The Yankee Atomic LOCA Manager was under the impression that the BE SBLOCA methodology description had been submitted to the NRC by Maine Yankee for review.

In 1990 Maine Yankee initiated a service request with Yankee Atomic to perform a scoping analysis of the effect of reduced steam generator pressure on the licensing basis transients and accidents. Steam generator pressure was decreasing steadily due to fouling of the steam generator tubes. The Yankee Atomic LOCA Group was responsible for addressing the LOCA aspects of this issue. In order to be responsive to the customer's request, Yankee Atomic used the only available LOCA analysis tool at that time, the BE model, to assess the effect of reduced steam generator pressure on the analysis of record. Memorandum LOCA 91-04 dated January 25, 1991 documented the use of the BE SBLOCA model. In this memorandum the BE model is erroneously stated to be the "licensing basis SBLOCA analysis," misrepresenting the BE analysis. The LOCA Group Manager was apparently out of the office when this memorandum was signed off, and the Department Manager signed off on the approval. It is apparent that the LOCA Group staff was confused about what the actual SBLOCA licensing basis was at this time. The analysis of record at this time was the 1977 Combustion Engineering analysis.

The analysis documented in LOCA 91-04 was eventually transmitted by reference to Maine Yankee as part of two other memoranda (NED 91-18 dated January 28, 1991, and TAG-MY-92-035 dated May 29, 1992). The LOCA Group Manager did sign his approval on the TAG-MY-92-035 memorandum. It is noted that the SBLOCA discussion in this memorandum is a very small part of the technical content and there is no mention of the referenced analysis as using the BE model. In April 1992, Yankee Atomic noted on a service request form (92-42) related to the reduced steam generator pressure that the Appendix K SBLOCA analyses were nearing completion, and that Yankee Atomic would provide the reanalysis of lowered steam generator pressure using the RELAP5YA evaluation model to Maine Yankee when completed. However, the BE

SBLOCA analysis memorandum was referenced by Maine Yankee in a 50.59 evaluation dated April 12, 1993 as part of the justification for operation with reduced steam generator pressure. The Appendix K SBLOCA results were forwarded to Maine Yankee on April 12, 1993, along with a draft 50.59 for Maine Yankee's use. These results were not referenced by Maine Yankee until January 13, 1994, when the 50.59 was revised.

In the various memoranda sent to Maine Yankee by Yankee Atomic that applied the BE SBLOCA model there were never any statements to the effect that the model was not NRC-approved. Memorandum LOCA 91-04 actually stated the contrary, which was incorrect. This indicates a lack of administrative control and communication within the Yankee LOCA Group and within the Yankee Atomic Maine Yankee Project Group. It also indicates a failure to appreciate the disclosures that should be made regarding application of non-NRC approved LOCA analysis technology in responding to plant support requests. There was a period of time available to correct the situation when the results of non-NRC approved methods were found to have been released to the customer.

The Review Team disagrees with the NRC position that only approved Appendix K evaluation models can be used in performing some scoping safety evaluations, including input to 50.59 evaluations. Models such as BE models can be appropriately used provided that the application does not replace the analysis of record, and provided that the use of the analysis method is clearly stated and justified. If there is any doubt regarding the appropriateness of such an application, then the NRC should be consulted prior to implementing the results of the analysis. Yankee Atomic's failure in this situation was in incorrectly characterizing the BE SBLOCA analysis as the licensing basis analysis, and furthermore not stating that the analysis used non-NRC approved methods and had restrictions on its use.

The Review Team notes that the effect of reduced steam generator pressure on the SBLOCA results for the magnitude of steam generator tube fouling and plugging that was being evaluated would not be expected to be significant. This is particularly true given that the analyses of record showed the SBLOCA PCTs to be lower than the LBLOCA PCTs. An evaluation could have been justified without any SBLOCA analysis. An evaluation could also have been justified using the BE analysis methodology provided that sufficient qualification was included, and provided that the analysis of record was not replaced.

V. ATTACHMENTS

- Technical Review Plan

TECHNICAL REVIEW PLAN

OBJECTIVE:

The purpose of this team is to provide an independent technical review of Yankee Atomic's small-break LOCA evaluation model (RELAP5YA SBLOCA) for Maine Yankee to determine compliance with 10CFR50.46 and 10CFR50 Appendix K requirements. If non-compliance issues are identified, the team will assess Yankee Atomic's cognizance of the problems, adequacy of the corrective actions taken, and communication of these issues with Maine Yankee and the NRC.

The results of this review will be presented to Duke Engineering & Services, Inc./Yankee Atomic to assist in formulating their response to the NRC's December 19, 1997 Demand For Information Letter. The Review Team will specifically look at inadequacies discussed in the NRC letter. Based on technical reviews and interviews with key Yankee Atomic staff, the team should arrive at some conclusion as to whether the technical issues (if present) were the result of personnel errors, process breakdowns, or failures within the quality verification process.

ON-SITE DATES: February 2 - 6, 1998
ENTRANCE MEETING: February 2
EXIT MEETING: February 6

REVIEW TEAM:

Z. L. Taylor*	Duke Power Company - Nuclear Assessment & Issues Division
G. B. Swindlehurst	Duke Power Company - Safety Analysis (Nuclear Engineering Division)
B. M. Dunn	Framatome Technologies
L. E. Hochreiter	Pennsylvania State University

* Team Leader

DETAILS:

In a Demand For Information letter addressed to Duke Engineering & Services, Inc./Yankee Atomic Electric dated December 19, 1997, the Nuclear Regulatory Commission states that engineering services provided to Maine Yankee for small-break LOCA analysis were not in compliance with regulatory requirements. The NRC also states that the Yankee Atomic, acting as a vendor for Maine Yankee, knowingly performed the inadequate analyses. This technical review was chartered to provide an independent review of the adequacy and compliance with 10CFR50.46 and 10CFR50 Appendix K of the subject evaluation model at the time of origination, evaluate the cognizance of Yankee Atomic personnel of the model weaknesses (if present), appropriateness and timeliness of

corrective actions that may have been taken, and whether or not the model weaknesses (if present) were effectively communicated to Maine Yankee and the NRC.

CONDUCT:

An entrance meeting will be held with appropriate DE&S management to discuss the Review Team's activities.

The review will consist of a technical review of the evaluation model used for the Maine Yankee small-break LOCA analysis. Interviews will be conducted with key personnel involved in the preparation, review and approval of the evaluation model. Communication with Maine Yankee and the NRC of any potential model weaknesses will also be assessed.

At the conclusion of the review and prior to departure, the team will hold an exit meeting with appropriate DE&S management to discuss the results of the review. A final report will be prepared and submitted to DE&S within two weeks following the exit meeting.

APPENDIX D

PERSONNEL BEHAVIOR ASSESSMENT

REPORT TO
DUKE ENGINEERING & SERVICES, INC.
ON ALLEGATIONS OF WILLFULNESS
RELATED TO THE
U.S. NUCLEAR REGULATORY COMMISSION
DECEMBER 19, 1997
DEMAND FOR INFORMATION

Winston & Strawn

J. Michael McGarry, III
Mark J. Wetterhahn

February 25, 1998

APPENDIX D

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REPORT TO DUKE ENGINEERING & SERVICES, INC. ON ALLEGATIONS OF WILLFULNESS RELATED TO THE U.S. REGULATORY COMMISSION DECEMBER 19, 1997 DEMAND FOR INFORMATION

I. EXECUTIVE SUMMARY

Winston & Strawn was requested to determine whether certain allegedly inadequate engineering analyses and materially incomplete and inaccurate information relating to the SBLOCA analyses used by Maine Yankee Atomic Power Company to demonstrate compliance with 10CFR50.46 was the result of willfulness, either deliberateness or careless disregard, on the part of personnel of Yankee Atomic Electric Company ("Yankee Atomic"), many of whom recently became part of Duke Engineering and Services, Inc. ("DE&S"). To respond to the request, Winston & Strawn reviewed the NRC's December 19, 1997 Demand for Information ("Demand"), reviewed relevant documents and interviewed a number of witnesses. These actions were taken in conjunction with an expert Technical Review Team whose function was to determine the technical adequacy of the SBLOCA analyses in the areas raised by the NRC.

Willfulness, as used by the NRC, embraces a spectrum of actions ranging from deliberate intent to violate or falsify to and including careless disregard for NRC requirements. Deliberate misconduct is defined as "an intentional act or omission that the person knows would cause a licensee to be in violation of any regulation or other NRC requirement." On the other hand, a finding of careless disregard indicates that the person acted with reckless indifference to a requirement or with disregard or utter unconcern of the consequences of whether there was compliance. The existence of a reasonable justification for an action would defeat a charge of willfulness despite the fact that the action was ultimately found to violate NRC requirements.

We found no actions on the part of any individuals associated with the specific issues contained in the Demand that would involve deliberateness. We found all individuals to be open, honest and communicative to us. We found no specific intent to violate any NRC regulation. Our evaluation, therefore, focused on whether there existed careless disregard for Commission requirements.

There were four specific allegations in the Demand, the first two of which had a common factual underpinning. The NRC alleged that because not all points of the SBLOCA spectrum could be reliably calculated by the code used by Yankee Atomic for the Maine Yankee facility, the requirements of 10CFR50.46 were not met. The Technical Review Team concluded that the standard industry practice, as utilized by experts in the LOCA field, was that the code should have the capability of analyzing all points within the prescribed spectrum. Yankee Atomic had taken the position that the identification of the limiting break, combined with a sufficient understanding of the physical phenomena which were occurring over the entire small break region, provided compliance with 10CFR50.46. The Technical Review Team recognized that NRC expectations regarding 10CFR50.46 were not completely documented, but rather had been

communicated to the LOCA community through its interactions with the NRC. Utilizing the appropriate legal standard, and bearing in mind that the Yankee Atomic LOCA Group was isolated and not part of the LOCA community, it was determined that the applicable regulation could be read as it was by Yankee Atomic. Actions to implement the LOCA Group's interpretation of the regulation did not evidence reckless indifference and, thus, no careless disregard was found.

With regard to the second issue, the NRC asserted that Yankee Atomic caused a violation of 10CFR50.9 in that Yankee Atomic provided inadequate information to Maine Yankee regarding the SBLOCA which did not reveal the code inadequacies discussed above. Many of the same considerations apply to this issue as to the previous one. In addition, the Technical Review Team found that on the whole the document submitted to Maine Yankee was sufficiently complete and accurate when judged using the perspective of its intended audience, i.e., one knowledgeable in the field, such as an NRC reviewer. Under the circumstances, we found that careless disregard of the regulations was not present.

The third issue relates to an assertion by the NRC that Maine Yankee had not provided a technical basis for one element of the SBLOCA analysis, the loss coefficient for the split downcomer nodalization, and, as a result, there was overprediction of core cooling and overstatement of the conservatism of the model. The Technical Review Team determined that the modeling approach utilized was reasonable and consistent with industry experience. The Technical Review Team determined that a deficiency in the quality assurance of a confirmatory calculation existed, but there was a reasonable explanation as to why it occurred, and that the evaluation undertaken by Yankee Atomic was appropriate. No inadequate analysis existed and the issue of deliberateness or careless disregard did not arise.

With regard to the fourth and last issue, the NRC asserted that Yankee Atomic memoranda used an unacceptable Best Estimate model rather than an approved Evaluation Model in evaluating the effect of a decrease in steam generator pressure on the peak clad temperature in the small break region when it should have known that such analysis would form the basis of a 10CFR50.59 analysis. The Technical Review Team determined that methods other than the approved Evaluation Model could have been utilized for the work undertaken. However, the Technical Review Team felt that limitations on the use of such methods should be stated. Here certain of the memoranda mischaracterized the Best Estimate model as the approved 10CFR50.46 Evaluation Model for the facility.

With regard to the use of the memoranda, we determined that it was understandable to have failed to contemplate that such work products would be used in a 50.59 analysis inasmuch as they either represented scoping calculations and/or were not thought at the time to be addressing design basis issues. However, it is also understandable that a Maine Yankee employee not expert in LOCA analyses would use these memoranda in performing the assigned 50.59 analyses. In evaluating whether careless disregard had occurred, we noted that the issue of whether decreased steam generator pressure impacted the design basis did not mature until late 1992 (i.e., some months after receipt of the last memorandum), as evidenced by the Maine Yankee

NRC Resident Inspector's suggestion that a 10CFR50.59 analysis should be performed. We found that the appreciation of the necessity for evaluating degraded conditions pursuant to 10CFR50.59 was in a state of transition at the time, with the NRC publishing Generic Letter 91-18 in late 1991. We concluded that while a misstatement was made that the Best Estimate model was the licensing basis SBLOCA analysis and while it was used without prior discussion with the NRC, the actions of Yankee Atomic personnel did not meet the test for careless disregard of the regulations in that we could not conclude that there existed a reckless disregard or careless indifference towards its responsibilities or the consequences of the actions taken under the circumstances assumed. Indeed, on the very date Maine Yankee's 10CFR50.59 analysis was internally approved, YAEC furnished Maine Yankee a draft 50.59 analysis on the precise subject based on its RELAP5YA Appendix K Evaluation Model.

In conclusion, we determined that, while in certain instances there may have been inadequate analysis associated with the SBLOCA analysis, there was neither deliberateness nor careless disregard resulting from the deficiencies discussed in the Demand. We found no willfulness on the part of the two individuals or any other Yankee Atomic personnel. We therefore believe that the Manager and Lead Engineer are capable of conducting their activities in the future in conformance with NRC requirements. We conclude that there was nothing developed as a result of our investigation on the conduct of Yankee Atomic and/or DE&S personnel that would prevent the improvements that we understand are being made from being successful and resulting in DE&S' activities being conducted in full compliance with NRC requirements and expectations.

II. INTRODUCTION

On December 19, 1997, the U.S. Nuclear Regulatory Commission ("NRC") sent Duke Engineering and Services, Inc. ("DE&S") and Yankee Atomic Electric Company ("YAEC" or "Yankee Atomic") a Demand for Information ("Demand") concerning the provision of "inadequate engineering analyses and materially incomplete and inaccurate information to an NRC licensee," namely Maine Yankee Atomic Power Company ("Maine Yankee" or "MYAPCo"). The subject matter of the Demand was a Loss-of-Coolant Accident ("LOCA") analysis used by MYAPCo to demonstrate compliance with 10CFR50.46 and Appendix K to 10CFR50. Specifically:

> YAEC prepared the small-break LOCA analysis which was utilized by MYAPCO during its operating Cycle 14 and to support its reload analyses for operating Cycles 14 and 15. See YAEC-1868. "Maine Yankee Small Break LOCA Analysis" (RELAP5YA SBLOCA analysis). YAEC also prepared the large-break LOCA analysis utilized by MYAPCO for Cycles 14 and 15. See YAEC-1160. "Application of Yankee WREM-BASED Generic PWR ECCS Evaluation Model to Maine Yankee" (WREM LBLOCA analysis).[1]

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[1] See Letter of December 19, 1997, transmitting Demand at 1.

The Demand required DE&S to submit responses to the NRC within 30 days, which date was extended pursuant to DE&S' request to February 27, 1998. Among the matters raised in the Demand was an NRC request for "[a]n explanation why the NRC should not consider the inadequate analyses, which apparently caused MYAPCo to be in violation of NRC requirements, to be the result of willfulness, either deliberateness or careless disregard, on the part of YAEC and/or DE&S personnel."[2] DE&S had purchased part of Yankee Atomic in late 1997, including the LOCA Group which was the focus of the Demand.

Winston & Strawn was requested by DE&S to address the willfulness aspect of the NRC's Demand. In response to this request, two attorneys from Winston & Strawn, J. Michael McGarry and Mark J. Wetterhahn (the "Willfulness Review Team"), both with a background in nuclear energy law, worked closely with the Technical Review Team chartered by DE&S to report on the technical issues contained in the Demand. The Technical Review Team consisted of experts in the field of LOCA analysis and the other subjects contained in the Demand.

The instant report describes the investigative process, the legal standards utilized, the determination as to willfulness and the bases for our findings for each of the four specific issues contained in Section III.A-D of the Demand, and the Willfulness Review Team's conclusion as to the ultimate issue contained in Section V.B of the Demand.

III. INVESTIGATIVE AND DELIBERATIVE PROCESS

At the outset, the Willfulness Review Team reviewed the Demand and associated documents to better understand the NRC's perspective, to identify the documents the NRC relied upon and to define the factual statements attributed to Yankee Atomic personnel. Additional relevant documents were also reviewed on a selective basis.

Because a determination of willfulness involves a mixed question of fact and law, it was necessary to rely upon determinations of fact made by the Technical Review Team. For example, the premise of NRC question V.B is that there was one or more "inadequate analyses" which caused a violation of NRC requirements which, in turn, resulted from willfulness on the part of YAEC and/or DE&S personnel. If the analyses discussed in Section III.A-D were determined in whole or in part to be technically adequate as judged by the appropriate standards, then the need for a willfulness determination as to that element is obviated. However, even should a determination be made that any of the analyses discussed in Section III of the Demand was in one or more ways inadequate, the deliberations and recommendations of the expert panel are necessary to determine the nature and extent of that deficiency from a technical standpoint and in relation to the NRC regulations.

As a result, the Willfulness Review Team closely coordinated its efforts with the Technical Review Team. In conjunction with the Technical Review Team, a number of questions were developed in order to assure that the Demand was completely analyzed and each of the issues,

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[2] Demand at 18.

premises and facts contained therein was questioned. The Willfulness Review Team attended the pre-interview meeting of the Technical Review Team. We were also present at all the interviews conducted by the Technical Review Team, including those of the Manager and Lead Engineer of the LOCA Group. Generally, during those interviews, the Willfulness Review Team elicited background information and conducted some follow-up questioning directed to matters contained in this report; the bulk of the questioning was by the members of the Technical Review Team.

Interviews and ensuing discussions took place over the course of four days. During the meetings, the Willfulness Review Team observed and participated in the deliberations of the Technical Review Team. Utilizing the questions previously generated as a guide, we challenged the Technical Review Team's conclusions to assure that they were understandable, defensible, and in a form that could be readily understood by an informed person and thus could be utilized as input to our deliberations.[3]

IV. LEGAL ANALYSIS OF WILLFULNESS

In analyzing the question of willfulness, we used NRC guidance and precedents and judicial analyses in formulating a series of questions to guide our deliberations. The term "willfulness," as used by the NRC, "embraces a spectrum of violations ranging from deliberate intent to violate or falsify to and including careless disregard for requirements."[4] Thus, at one end of the "willfulness" spectrum are violations involving a "deliberate intent to violate or falsify."[5] At the same end of the willful spectrum is 10CFR50.5 (1997), the NRC deliberate misconduct rule, which prohibits deliberate violations of NRC regulations and deliberate submittals of inaccurate material information to the NRC. For purposes of the deliberate misconduct rule, "deliberate misconduct" is defined as "an intentional act or omission that the person knows . . . [w]ould cause a licensee to be in violation of any rule, regulation," or other NRC requirement. Thus, for a violation of an NRC regulation to be "deliberate," there must be both an intent to take a certain action and knowing intent to violate an applicable requirement.

Examples of deliberate behavior, as previously cited by the NRC, include: a security supervisor reteaching exam materials after several guards answered a question incorrectly, allowing the

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[3] We have reviewed the report of the Technical Review Team entitled "Yankee Atomic SBLOCA Technical Review Report ("Technical Review Report"). While certain portions of that report are cited herein, we have based our findings on the Technical Review Team's report in its entirety. A number of factual statements in this report are not contained in the Technical Review Report. These statements were based upon interviews conducted and information received.

[4] NRC Enforcement Policy IV (C), 60 Fed. Reg. 34,381, 34,385 (1995).

[5] Id.

guards to change their answers, then submitting the guards' revised answers for grading;[6] making false statements on an application for access authorization;[7] and a contractor providing false information about the capability of its product. These cases are examples of actions by individuals with specific knowledge that their actions violated NRC regulations.

Deliberate behavior is, however, not the only standard applied to prove willfulness. At the other end of the spectrum of "willful" violations are those that result from careless disregard. "Careless disregard" is not defined in NRC regulations; however, a clear pronouncement by the NRC Staff regarding "careless disregard" was provided in the context of proposing the deliberate misconduct rule:

> Careless disregard has been described as a showing of disregard for a governing statute or an indifference to its requirements . . . . A finding of careless disregard indicates that the person acted with reckless indifference to the requirement, or with disregard (or utter unconcern) of the consequences or whether there was compliance. This recklessness involves, at a minimum, an unconcern as to whether a requirement was or will be violated, or a situation in which an individual blinds himself or herself to the realities of whether a violation has occurred or will occur.[8]

The NRC's description of "careless disregard" is similar to language that has been accepted by the Supreme Court. The Supreme Court has also held that if an individual makes a reasonable, good faith effort to determine what constitutes a violation of the law, then he cannot be acting with careless, or reckless, disregard. Accordingly, a willful violation cannot result if a licensee has considered NRC requirements and reached the erroneous conclusion that its actions will not violate relevant statutory or regulatory provisions. The existence of a licensee's reasonable justification for its actions should, therefore, defeat a charge of willfulness, despite the fact that the licensee undertook an action that was ultimately found to violate NRC requirements.

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6 Pennsylvania Power & Light Co., EA 94-212 (May 9, 1995) (issuing a Severity Level III violation to the licensee for violating 10CFR50.9(a)); Darryl R. Zdanavage, IA-95-011 (May 9, 1995) (issuing a Severity Level III violation to the exam proctor for violating 10CFR50.5(a)(2)).

7 E.g., Juan Guzman, IA 96-018 (Apr. 19, 1996); Gerald O. Eckard, IA 94-016 (Aug. 3, 1994).

8 55 Fed. Reg. 12,374, 12,375 (Apr. 3, 1990) (citations omitted). See also 52 Fed. Reg. 49,362, 49,365 (Dec. 31, 1987) ("The concept of 'careless disregard' goes beyond simple negligence . . . [it] connotes a reckless disregard or callous . . . indifference toward one's responsibilities or the consequences of one's actions.") (citations omitted). In sum,"careless disregard" does not require a conscious decision to violate a known requirement; "careless disregard" requires that a violator act with indifference (beyond mere negligence) to the applicable requirement.

V. FINDINGS

The findings on the individual issues outlined in Section III of the Demand and the actions of the individuals identified therein must be viewed in the context of the history of development of the SBLOCA code, RELAP5YA, by Yankee Atomic for Maine Yankee. In addition, key decisions made by individuals other than the persons named in the Demand significantly affected the actions of the named individuals, the alternatives available to them, and their decision-making process. Perhaps equally important is an understanding of the internal reporting relationships, responsibilities, and philosophy of YAEC and its relationship with Maine Yankee.

In response to a decision by YAEC management to broaden their analytical capabilities to support their customers, a decision was made to initiate the RELAP5YA/MOD1 development project in 1980 to analyze loss-of-coolant accidents in accordance with the requirements of 10CFR50.46 and Appendix K to 10CFR50. The RELAP Code was originally developed by INEL. In 1983, YAEC submitted a request to the NRC for approval of RELAP5YA as the code to meet the requirements of II.K.3.30 of NUREG-0737 related to post-TMI issues.[9]

While the NRC review was in progress, Yankee Atomic applied the RELAP5YA code to the resolution of an issue for Maine Yankee raised by the NRC related to reactor coolant pump trips. During this time frame, it was recognized that there were some inherent difficulties in the RELAP5YA code which would require significant work to overcome. Certain individuals within the LOCA Group, including the Manager and Lead Engineer, although the former did not hold a management position at the time, requested that YAEC management permit them to develop a new code without the same inherent limitations as the code then under review by the NRC. Management decided to proceed with the use of RELAP5YA. While the full details and the wisdom of such a management decision are well beyond this review, it indicates the mindset of the LOCA Group that they were to use RELAP5YA and overcome its difficulties in meeting the technical requirements of the plants for which they provided a SBLOCA Evaluation Model.

Also as determined by the Technical Review Team, the process by which approval of the RELAP5YA code was sought from the NRC in 1983, and which extended over a period of six years, was flawed in that in parallel with the approval of the Evaluation Model, plant-specific applications were not performed. This resulted in the approval of a generic code which was not significantly challenged and whose application to a specific reactor, e.g., Maine Yankee, was unproven, significantly lengthening the time necessary to obtain a final specific analysis which could be implemented for Maine Yankee.

In our view, these factors were an influence on the LOCA Group to make RELAP5YA work for Maine Yankee and increased the risk to both Maine Yankee and YAEC should that code not be demonstrated to comply with the requirements of 10CFR50.46. These critical decisions were made prior to the individual designated as Manager in the Demand being the incumbent in the position; he and the LOCA Group were saddled with these handicaps.

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[9] Id. at III.B.

It was also clear that the YAEC LOCA Group was working in relative isolation from the LOCA community which was dominated by the nuclear steam supply system vendors and those companies selling core reloads. The Yankee Atomic LOCA Group was relatively small and, although all individuals interviewed appeared to be knowledgeable about SBLOCA computer models and phenomena and are technically qualified,[10] they were challenged by resource limitations. Yankee Atomic management did not assure sufficient experience and depth in the staff of the RELAP5YA project team, in particular at the beginning of the project in the early 1980's. The Technical Review Team expressed its view that "a LOCA development program requires a very significant resource commitment which includes a sustained critical mass of expertise and continuity of key personnel."[11]

After the completion of the NRC's generic approval of RELAP5YA in 1989, a situation arose which led to some of the later deficiencies. During the generic RELAP5YA approval process, the LOCA Group was able to communicate directly with the NRC in furtherance of the licensing of that code; thereafter, application of the generic code to Maine Yankee became a Maine Yankee licensing matter which led Yankee Atomic personnel, including those in the LOCA Group, to believe that they could have no direct interface with the NRC.[12]

The structure preordained by Maine Yankee and Yankee Atomic management also appeared to inhibit the free flow of communications between the Yankee Atomic LOCA Group and their ultimate customers at Maine Yankee, although Maine Yankee personnel appear to have been briefed periodically on the difficulties that the Yankee Atomic LOCA Group were encountering in implementing RELAP5YA for Maine Yankee. Formal communications from the LOCA Group were funneled to a Maine Yankee project group at Yankee Atomic and from there to a single point of contact at Maine Yankee to be disseminated to the appropriate Maine Yankee organization. Submittals intended for the NRC were made by a Maine Yankee licensing group which seemingly had little contact with the Yankee Atomic LOCA Group. While there may have been legitimate reasons which resulted in such lines of communications (which reasons we did not explore and did not consider to be relevant to our inquiry), the fact is that communications were less than ideal and, in our view, contributed to the difficulty in resolving the issues discussed in the Demand.

Discussions with Yankee Atomic personnel revealed to us a rather uneven understanding of the NRC requirements related to design basis, the requirements of 10CFR50.59, and of the term "safety-related." Inasmuch as these events occurred in the late 1980s and up to 1993, we attempted to judge this lack of appreciation and the resulting actions by YAEC employees by contemporaneous standards. Clearly, the emphasis with regard to meeting NRC standards was to assure adequate safety as opposed to today's focus on compliance. It was also clear that the

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10 Id. at III.D.4.

11 Id.

12 Id. at III.D.6.

employees of the LOCA Group considered themselves a technical resource and not part of the direct support of Maine Yankee operations. The LOCA Group was largely isolated and had minimal experience in operations: their entire emphasis was on technical development of information that could be used in the implementation of 10CFR50.46 and Appendix K to 10CFR50.

With this background, we turn to the issues posited in the Demand for Information, Section III.A-D.

VI. RESPONSE TO SECTION III OF THE DEMAND

A. Deliberateness

It is our conclusion, as bolstered by the opinion of the Technical Review Team, that there was no deliberateness revealed in our investigation on the part of any YAEC or DE&S employee or, for that matter, any other person within the scope of the issues covered by the Demand. We found all individuals that we interviewed to be open, honest, and cooperative. If anything, they appeared to be overwhelmed by the situation before them, and they still lacked an appreciation as to how matters that they perceived to be appropriate technical decisions -- or, at the most, technical differences among experts in their specialty -- had turned into an issue of willfulness in which their veracity was being challenged.

We did not encounter a situation where individuals tried to shift the blame to others; all were willing to defend what they believed to be valid technical decision-making within the scope of the NRC regulations as they understood and interpreted them. Thus, we have determined that to the extent that there were inadequate analyses as discussed below and in the report of the Technical Review Team, which could have resulted in Maine Yankee being in violation of NRC requirements, they were not the result of deliberateness on the part of YAEC or DE&S personnel. We have no indication of any individuals proceeding despite knowledge that his or her actions were wrong. Thus we conclude, using the standards set forth above, that no deliberateness was involved.

B. Response to Sections III.A and B

We turn now to a discussion of the four issues contained in Section III of the Demand to determine whether there was careless disregard on the part of YAEC or DE&S personnel which could have caused Maine Yankee to be in violation of NRC requirements. In doing so, we utilize the Technical Review Team's findings as to whether there existed inadequate analyses in the specific areas delineated in Section III.A-D.

A. ***During Cycle 14 operations, and in the Cycle 14 and Cycle 15 reload analyses, Maine Yankee used apparently unacceptable Evaluation Models which could not calculate or reliably calculate ECCS performance.***

B. ***MYAPCo maintained information and submitted to the NRC Core Performance Analysis Reports, in support of Cycle 14 and Cycle 15 reload applications, which apparently were not complete and accurate in all material aspects.***

We treat subsections A and B of Section III of the Demand together since they are largely based upon common factual issues. Our inquiry was bounded by the small break analyses, i.e., the small break spectrum up to and including a break size of approximately 0.6 ft.^{2}. The NRC contends that in order for LOCA codes to be acceptable, they must not only be capable of calculating any point on the break spectrum but must be capable of producing reliable calculations. The NRC asserts that RELAP5YA, which is the code for evaluating the portion of the break spectrum up to 0.6 ft.^{2}, was not capable of calculating break sizes of and greater than 0.35 ft.^{2}. The NRC reasons that if it was not possible to analyze any point on the break spectrum, it was not possible to confirm that the limiting break within the spectrum had been identified. The NRC asserts that both the Manager and Lead Engineer were significant contributors to the preparation of the RELAP5YA code and should have recognized it did not meet the requirements of the regulations as interpreted by the NRC.

For their part, the Manager and Lead Engineer assert that they believed their implementation of the RELAP5YA code, as it applied to Maine Yankee, to be in compliance with the Commission's regulations and, in particular, 10CFR50.46 and Appendix K. They reached this conclusion based upon the successful running of the RELAP5YA code at a number of break sizes smaller than 0.35 ft.^{2} and at 0.35 ft.^{2} until the code terminated. They asserted that they had identified the limiting break in the small break spectrum as being at 0.15 ft.^{2} and had done sufficient calculations and had a sufficient understanding of the physical phenomena which were occurring in the small break region, i.e., up to 0.6 ft.^{2}, such that they could tell with reasonable assurance that a limiting break would not occur at 0.35 ft.^{2} or greater. They stated that the RELAP5YA code had not "failed" in the way alleged by the NRC, but it had terminated and had not been run further either at 0.35 ft.^{2} or larger because it was not necessary. The Manager and Lead Engineer were of the view that the instabilities and oscillations which had been increasing as 0.35 ft.^{2} was approached, were conservatively accounted for in determining the Peak Clad Temperature ("PCT")[13] and would not have affected the ability of the code to reliably determine that the limiting break size for the small break spectrum had already been identified at smaller break sizes. They stated that the

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13 We use the term PCT to mean the highest temperature of the fuel cladding reached for a particular break size and also to mean the highest fuel clad temperature for any assumed break size in the spectrum, usually modified by the word "limiting."

small break size did not contribute the limiting PCT which was in the large break spectrum covered by another code.[14]

Our evaluation starts with 50.46(a)(1)(i). In pertinent part, that section states:

> ECCS cooling performance must be calculated in accordance with an acceptable evaluation model and must be calculated for a number of postulated loss-of-coolant accidents of different sizes, locations, and other properties sufficient to provide assurance that the most severe postulated loss-of-coolant accidents are calculated . . . . [T]he evaluation model must include sufficient supporting justification to show that the analytical technique realistically describes the behavior of the reactor system during a loss-of-coolant accident.

The Technical Review Team spent considerable time in discussing the proper interpretation of this portion of 50.46. Based upon its members' experience in implementing the cited section, they viewed that the industry standard practice is that an Appendix K SBLOCA evaluation model be capable of analyzing any break size within the plant's SBLOCA licensing basis. That is not to say that the calculation model had to have been run for each point or that there could not be a point within the spectrum that the model code could not be capable of generating an analytical value, but, simply, that the capability should exist. An example discussed during the Technical Review Team's deliberations was that if a code could reliably calculate peak clad temperatures at 0.3 $ft.^2$ and 0.4 $ft.^2$, the fact that the code was somehow incapable of performing a calculation at 0.35 $ft.^2$ would not be a deficiency if the limiting break had been identified and it was understood why the code was incapable of calculating a value at that intermediate point. Based upon discussions with the individuals involved, it was the Technical Review Team's judgment that the RELAP5YA Evaluation Model has not demonstrated the capability to analyze the complete range of the historical Maine Yankee SBLOCA break spectrum.[15]

The Technical Review Team did state that Yankee Atomic's identification of the limiting PCT within the range of break sizes evaluated and conclusion that this was the limiting PCT for the small break spectrum based on the decreasing trend of PCT for smaller and larger break sizes, was consistent with Yankee Atomic's interpretation of 10CFR50.46 and Appendix K to 10CFR50. The Technical Review Team noted that in response to its questions, the LOCA Group provided additional verbal justification consisting of an explanation of SBLOCA phenomena which supported its conclusion. The Technical Review Team found that additional analysis results from other codes tend to support the

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[14] Id.

[15] Id.

Yankee Atomic position, although some of the trends are not consistent with that position.[16]

Starting with the hypothesis that the course taken by the LOCA Group deviated from the industry standard practice to calculate PCTs over the entire small break spectrum, we examined the regulation to determine whether Yankee Atomic's interpretation, i.e., the focus of the regulation was the proper identification of the limiting break size and limiting PCT, evidenced reckless indifference or utter unconcern of whether there was compliance with the regulation. We believe that one reading of 10CFR50.46 could lead to the LOCA Group's interpretation. Section 50.46 requires that "ECCS cooling performance must be calculated in accordance with an acceptable evaluation model and must be calculated for a number of postulated, loss-of-coolant accidents of different sizes, locations and other properties sufficient to provide assurance that the most severe postulated loss-of-coolant accidents are calculated." (Emphasis supplied.) The focus of the LOCA Group was in providing sufficient assurance that the limiting break had been identified; this is clearly the intent of the cited regulation. By its terms, the regulation requires that the PCT for "a number" of sizes be calculated. With the limiting PCT identified, the LOCA Group believed that running the remainder of the break spectrum was unnecessary since they had confidence they had determined that the most severe break in the small break spectrum had been identified based upon an understanding of the physical phenomena involved.

The Technical Review Team noted that based on a broader knowledge of SBLOCA phenomena and results from other codes, Yankee Atomic was confident that SBLOCAs were bounded by LBLOCAs and based on this expectation, Yankee Atomic accepted the results from RELAP5YA as adequate for showing compliance with the regulations. The Technical Review Team stated its understanding that this may have been a correct conclusion, but because it was based, in part, on information beyond demonstrated results of runs of the RELAP5YA code for Maine Yankee, the situation should have been communicated to the NRC prior to implementation. The Technical Review Team concluded that a number of issues related to RELAP5YA implementation (as well as a number of other issues related to the Demand) should have been communicated to the NRC prior to implementation. Clearly, in hindsight, that is the case.

However, we understand the situation in which the LOCA Group found itself was not entirely of its own making and have taken this into account in our review. Maine Yankee had been told that no submittal to the NRC to satisfy TMI Item II.K.3.31 was necessary. Communications with the NRC were understood by the LOCA Group to be discouraged. Therefore, they did not believe they could informally communicate with the NRC. A corporate culture stressing independence, poor management oversight over the project, the LOCA Group's isolation from the remainder of the entities developing

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16 Id.

LOCA codes, interactions with Maine Yankee, and actions by the NRC also contributed to difficulties in implementing the RELAP5YA Code as applied to Maine Yankee.[17]

With regard to our analysis of careless disregard, we considered the following elements:

1. The complete small break spectrum had not been analyzed.

2. The Technical Review Team had questions as to the behavior of the PCT in the unanalyzed region based upon code behavior and previous results with other codes.

3. The decision that all points on the complete small break spectrum did not have to be run came after the LOCA Group was unsuccessful in completing the run at 0.35 ft.^{2} after significant effort over a number of years of working with RELAP5YA.

4. RELAP5YA was not run for a break size of 0.5 ft.^{2}, the limiting small break size for the Appendix K analysis of record.

5. It was a priority of the LOCA Group to complete the project.

6. If the LOCA Group had inquired of the NRC or determined its expectations from others, it could have avoided the situation.

7. On its face, 10CFR50.46 would not prohibit the combination of analysis and technical justification based upon an understanding of physical phenomena.

8. NRC expectations regarding 10CFR50.46 were not completely documented, but rather were communicated to the LOCA community through its interactions with the NRC.

9. The LOCA Group believed that the successful running of the RELAP5YA code at a number of break sizes smaller than 0.35 ft.^{2} and at 0.35 ft.^{2} until the code terminated, combined with a sufficient understanding of the physical phenomena which were occurring in the small break region, i.e., up to 0.6 ft.^{2}, and an understanding of the differences between RELAP5YA and the code of record was sufficient to comply with 10CFR50.46.

10. Members of the LOCA Group were able to provide the Technical Review Team justification which supported their code. The Technical Review Team found that such results tend to support the Yankee Atomic position although some trends do not.

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[17] Id. at III.D.1.

11. Although it was unable to draw a definitive conclusion regarding the PCTs for the unanalyzed portion of the Maine Yankee SBLOCA spectrum, the Technical Review Team concluded that the SBLOCA PCTs for all the analyses reviewed met the 10CFR50.46 2200°F criterion and that SBLOCAs remain bounded by LBLOCAs.

12. A combination of longstanding institutional issues, isolation from the LOCA community, poor management, inadequate support within the Company, interactions with Maine Yankee, and actions by the NRC placed roadblocks in the way of the successful completion of this project.

13. The technical capabilities of the LOCA Group were acceptable for performing LOCA analysis, and the project included extensive assessment of RELAP5YA to scaled test facility SBLOCA data.

Weighing all these factors, we find that there was a basis for the LOCA Group's interpretation of 10CFR50.46, albeit one in contrast to NRC and industry expectations; we also find there was a basis for the LOCA Group to conclude they had identified the limiting break in the small break spectrum based upon their knowledge of the phenomena associated with small breaks. We therefore conclude that a reasonable justification existed for the LOCA Group's actions, and thus we cannot conclude that there was reckless indifference to the requirements of 10CFR50.46 or utter unconcern of the consequences in that the LOCA Group thought it was complying with a permissible interpretation of 10CFR50.46. In these circumstances, we cannot say that an individual in the LOCA Group blinded himself or herself to the realities of whether a violation would occur. We conclude that careless disregard of the Commission's requirements was not involved with respect to the issues in Section III.A of the Demand.

Turning now to subsection B, the NRC asserted that Maine Yankee maintained information in its files (YAEC-1868) and submitted to the NRC Core Performance Analysis Reports ("CPAR") in support of operating Cycle 14 and Cycle 15 reload analyses which were not complete and accurate in all material respects in apparent violation of 10CFR50.9(a). The NRC further asserted that the Manager never brought the deficiencies to the attention of the cognizant Manager in charge of the Engineering Section of the Licensing and Engineering Group of Maine Yankee, who was the manager kept directly apprised by Yankee Atomic and by the Manager on the development of the specific RELAP5YA Evaluation Models.

For their part, the involved YAEC personnel maintain that they believed that they had complied with the requirements of 10CFR50.46(a)(1) and that the documents were complete and accurate in all material respects and they adequately communicated with Maine Yankee personnel.[18]

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[18] Our investigation did not reach the question whether individuals at Maine Yankee may have been knowledgeable in the requirements of 10CFR50.46 and RELAP5YA. See n.59,

It is clear that the Yankee Atomic LOCA Group expected that YAEC was subject to audit by the NRC with regard to the application of the Evaluation Model to Maine Yankee and its compliance with Commission regulations. We believe that this belief was reasonable, particularly in light of correspondence between Maine Yankee and the NRC that such was contemplated.[19]

The Technical Review Team concluded that YAEC-1868 documents the results of the SBLOCA analysis performed by Yankee Atomic using the NRC-approved YAEC-1300P SBLOCA Evaluation Model. It further concluded that the document was sufficiently complete and accurate as a summary of the SBLOCA calculations that were performed. It also found that it was understandable to its intended audience and was suitable for a licensing submittal in support of Maine Yankee. The Technical Review Report concluded that the amount of technical information included was appropriate for any knowledgeable person to understand the results of the analysis. The Technical Review Team found there was no intended concealment of information which would have identified any non-compliance with NRC regulations. The Technical Review Team stated that YAEC-1868 could be understood by a knowledgeable engineer not trained in the LOCA licensing process to be complete and in compliance with the regulations. The Technical Review Team concluded that the compliance statement and the supporting analyses in YAEC-1868 would be understood by an NRC reviewer and would have led to interactions with Yankee Atomic.[20]

Our review of YAEC-1868 confirmed that there were instances which erroneously stated that the complete break spectrum was analyzed. However, we believe that the document should be read as a whole. The arguments advanced by the Manager and Lead Engineer as to RELAP5YA's compliance with 10CFR50.46 are discussed in the body of the document itself. Inasmuch as this document was maintained at Maine Yankee and subject to audit by the NRC, we believe, as does the Technical Review Team, that any reviewer from the NRC [21] sent to audit compliance with 10CFR50.46 would have

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infra. See also Technical Review Report at IV.B.

19 May 8, 1989 letter to C.D. Frizzle, President, Maine Yankee Atomic Power Company from Patrick M. Sears, Project Manager, NRC.

20 Id.

21 We believe that the standard advanced by the NRC to the effect that "language would not signify to an individual without expertise in LOCA code that RELAP5YA had failed and was not capable of calculating ECCS performance" (Demand at 11) is not the appropriate one for determining whether there had been careless disregard of the regulation. We believe the issue of concealment should be judged from the point of view of a cognizant NRC reviewer knowledgeable in computer codes used to satisfy 50.46. See Virginia Electric Power Company (North Anna Power Station, Units 1 and 2), CLI-76-22, 4 NRC

understood the compliance statements and supporting analyses and would have inquired further. Finally, we observe that the Technical Review Team felt that the document was suitable for licensing[22] which we understand to mean complete and accurate, subject to anticipated NRC questions and revision, as appropriate.

In addition to the considerations previously outlined in response to Section III.A of the Demand and in light of the above, we weighed the following considerations in determining whether careless disregard of NRC requirements existed related to Section III.B:

1. YAEC-1868 could be understood by a knowledgeable engineer not trained in the LOCA licensing process to be complete and in compliance with the regulations.

2. The Abstract of YAEC-1868 states that "[e]valuations were performed over a complete range of break sizes . . . ." Other sections also contain similar language.

3. YAEC-1868 was sufficiently complete and accurate as a summary of the SBLOCA calculations that were performed. The amount of technical information included was appropriate for any knowledgeable person to understand the results.

4. The scope of the analysis as contained within the report is characterized correctly and there is no concealment regarding the "failure" of the run at 0.35 $ft.^2$ and that no runs of larger break sizes within the small break spectrum were run.

5. But for the May 8, 1989 NRC letter, YAEC-1868 would have been submitted to the NRC.

6. Yankee Atomic had a reasonable belief that the NRC would perform an audit on YAEC-1868.

7. The statements in YAEC-1868 are consistent with the LOCA Group's understanding of the regulations.

8. The compliance statements and supporting analyses in YAEC-1868 would be understood by an NRC reviewer and would have led to interactions with Yankee Atomic.

480, 491 (1976). Such a reviewer or auditor would be able to appreciate the significance of the statements contained in YAEC-1868. The experts within the Technical Review Team were able to do so.

22 Technical Review Report at IV.B.

On this basis, we conclude that to the extent any deficiency existed, it did not represent reckless indifference to a requirement or utter unconcern whether there was compliance with the requirement. We cannot say that any individual blinded himself or herself to the realities of whether a violation occurred. Thus, no careless disregard was involved.

C. Response to Section III.C

C. During Cycle 14 operations and in the Cycle 14 and Cycle 15 CPAR, MYAPCo used an apparently unacceptable SBLOCA Evaluation Model which over-predicted core cooling.[23]

In this subsection, the NRC asserts that the LOCA Group misapplied the Alb-Chambre penetration correlation, which is an empirical calculation of the penetration factor of the injected ECCS water penetrating the downcomer annulus into the lower plenum. The NRC concludes that because the analysis did not provide a basis to assume full penetration of the emergency core cooling system injection, it provides no basis to derive the loss coefficient of 600 used for the split downcomer nodalization. The NRC further asserts that these deficiencies resulted in overprediction of core cooling and overstatement of the conservatism of the model. The NRC states that the Manager and Lead Engineer should have realized during the work associated with the RELAP5YA analysis, that the Alb-Chambre correlation had been incorrectly applied. It states that adequate QA review would have revealed the errors and the unacceptability of the RELAP5YA SBLOCA analysis described in YAEC-1868. The NRC ultimately states that based upon these errors, YAEC caused Maine Yankee to rely on an unacceptable SBLOCA Evaluation Model in violation of 10CFR50.46(a)(1).

The LOCA Group asserts that the NRC description does not reflect their reasoning concerning core penetration and the actions that they took. While admittedly there was an arithmetic error in the implementation of the Alb-Chambre correlation, the results as presented in the calculation would not cause a reviewer providing QA oversight to have recognized that the Alb-Chambre correlation had been mistakenly calculated. Furthermore, the LOCA Group states that even if the negative value of Alb-Chambre correlation had been discovered, a negative range, i.e., less than zero, would not be indicative of a meaningless result and merely would have caused them to further examine the values used in the correlation. The LOCA Group denies that such calculations also indicate other possible errors in application of the Alb-Chambre correlation.

The Technical Review Team concluded that the use of the Alb-Chambre correlation as a confirmation of the modeling approach which includes the junction loss coefficient of 600 in the reactor downcomer is reasonable given the available data and the deficiency in

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[23] It may be argued that Section III.C does not allege willfulness, but merely asserts that there were specific deficiencies in the LOCA analysis and related quality assurance errors. However, we have analyzed this issue in accordance with Section IV.B of the Demand.

the code. Early applications of the RELAP5YA SBLOCA Evaluation Model to Maine Yankee identified excessive ECCS bypass relative to what was expected based on scaled test facility data and the results of other codes. Eventually an artificially large loss coefficient was introduced in the junction connecting the two volumes representing a split reactor vessel lower downcomer. The value of this loss coefficient was varied to obtain a balance between the expected ECCS penetration and the effect on steam venting via the break. A value of 600 was selected as an appropriate value. The amount of ECCS penetration obtained with this modeling approach was justified, in part, by the Alb-Chambre correlation. This correlation was applied to confirm that the amount of ECCS penetration predicted by RELAP5YA was conservative. The Review Team considered this approach to be an acceptable compensation for a RELAP5YA deficiency, and the use of a value of 600 obtained an amount of ECCS penetration that was consistent with industry experience. The Technical Review Team concluded that this modeling approach is not expected to result in an overprediction of core cooling, but since the calculations were not completed for all break sizes, it cannot be definitively confirmed.[24]

An arithmetic error was made in the application of the Alb-Chambre correlation, but not identified during the quality assurance process. The Technical Review Team found this to be a failure of the quality assurance process, but was able to conclude there was a reasonable explanation as to why it occurred. The arithmetic error did not skew the result of the calculation, which was in the range of the expected result that complete penetration was predicted. The correlation can produce results in excess of the value of 1 (in this case, a value of 8) which have the meaning of complete ECCS penetration. The Technical Review Team found that a person performing a quality assurance review is influenced by the result based on experience and expectations and the errors encountered here are the most difficult errors to identify. A correct application of the Alb-Chambre correlation without the arithmetic error would have produced negative values indicating complete ECCS bypass. It was concluded that this result would have been immediately recognized by Yankee Atomic as non-physical for the SBLOCA conditions of interest. The cause of the non-physical result would have been traced to excessively conservative input values. More reasonable values would then be input to the correlation, and reasonable and valid results indicating significant ECCS penetration would have resulted. Therefore, the Technical Review Team concluded that although the application of the Alb-Chambre correlation contained an error, the modeling which incorporated the loss coefficient with a value of 600 remains valid and, thus, the results of the error in applying the Alb-Chambre correlation did not result in invalid input to the SBLOCA analyses.[25]

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[24] Id. at IV.C.

[25] Id.

The Technical Review Team concluded that the use of the Alb-Chambre correlation as a confirmation of the modeling approach which includes the junction loss coefficient of 600 in the reactor vessel downcomer was reasonable given the available data and a need to address a deficiency in the code. The Technical Review Team believed that had there been a submittal of the Maine Yankee SBLOCA application to the NRC, this modeling approach would have been discussed and reviewed. The Technical Review Team believed that this model could have been approved by the NRC in this form or with some revision.[26]

It is therefore our conclusion that there is no inadequate analyses involved in subsection III.C and, therefore, the issue of deliberateness or careless disregard does not arise. It is our view that the actions of the cited individuals were reasonable and there was no evidence of careless disregard.[27]

D. Response to Section III.D

D. MYAPCo used an apparently unacceptable Best Estimate RELAP5YA SBLOCA Evaluation Model to calculate ECCS performance.

The NRC asserts that Maine Yankee performed a safety evaluation in order to determine if a decrease in steam generator pressure involved an unreviewed safety question pursuant to the requirements of 50.59 and, in doing so, used an unacceptable Best Estimate ("BE") RELAP5YA Evaluation Model in evaluating small break ECCS LOCAs in apparent violation of 50.46(a)(1). The NRC states that Maine Yankee's March, 1993 50.59 analysis relied on an analysis of the effect of reduction in steam generator pressure on ECCS performance prepared by Yankee Atomic which used, among other analyses, the Best Estimate RELAP5YA SBLOCA analysis. This document was entitled LOCA 91-04 which was approved by the Manager of the Yankee Atomic nuclear engineering department for the Manager of the LOCA Group. The

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26 Id.

27 While not raised by the NRC in its Demand, the Technical Review Team discussed whether the loss coefficient changes required NRC approval inasmuch as it concluded that due to the non-physical value used and the significance of this input parameter, it is an important model change. While the Technical Review Team recognized that the loss coefficient was literally an input to the model, it concluded that in its opinion it was an important model change (Technical Review Report at III.D.1, IV.C). We note that the Technical Review Team recognized that had the Maine Yankee SBLOCA application been directly submitted to the NRC, this model "could have been approved by the NRC in this form or with some revision" (Id. at IV.C). Because of these factors and the confusion that existed due to the NRC's May 8, 1989 letter, we conclude that Yankee Atomic had a basis for assuming that the NRC would have reviewed the matter in due course and, therefore, would have acted consistent with the Technical Review Team's expectations.

Manager received a copy of a YAEC memorandum, Impact of Lower Steam Generator Pressure on the safety analysis, NED 91-18, which relied on LOCA 91-04 to evaluate the impact of reduced steam generator pressure on the small and LBLOCA analyses. A Yankee Atomic memorandum dated May 29, 1992 entitled Steam Generator Pressure and Heat Transfer Coefficient Performance (TAG-MY-92-035) concluded that "the lower initial steam generator pressure did not affect the results of the licensing analysis," which conclusion was based in part on NED 91-18. The May 29, 1992 memorandum states that it is "safety-related." The Manager approved the May 29, 1992 memorandum.

The NRC asserts that a 50.59 analysis cannot confirm that the ECCS performance will be adequate unless the 50.59 analysis uses LOCA Evaluation Models acceptable to demonstrate compliance with 50.46 and then reasons that the Best Estimate RELAP5YA model could not be utilized because in January, 1989, the NRC approval was for use of RELAP5YA as the Appendix K Evaluation Model for Maine Yankee. Additionally, the NRC states that there were some parts of the Best Estimate model which did not comply with Part 50, Appendix K. The NRC states that it is reasonable to conclude that the Manager of the Yankee Atomic LOCA Group was aware that the Best Estimate approach deviated from the approach approved by the NRC and that a Best Estimate RELAP5YA model would not be acceptable for use in licensing matters, without NRC approval. The NRC believes it is reasonable to conclude that the Manager knew that the analysis which Yankee Atomic had performed would be used by Maine Yankee in a 50.59 analysis or other safety analyses. The NRC concludes that in view of the intended use of the YAEC analysis, the Manager should have provided Maine Yankee with an analysis which met NRC requirements and that by providing to Maine Yankee an unacceptable analysis of the effects of reduction steam generator pressure, Yankee Atomic caused Maine Yankee to apparently violate 10CFR50.46(a) by relying on an unacceptable SBLOCA Evaluation Model to calculate ECCS cooling performance in preparing a 50.59 analysis.

There are two separate elements of our evaluation of this issue. The first relates to the Maine Yankee service requests which were issued in 1990. These asked Yankee Atomic to perform what the Technical Review Team found to be a "scoping analysis" of the effect of reduced steam generator pressure on the licensing basis transients and accidents. Steam generator pressure was decreasing steadily due to fouling of the steam generator tubes. The Yankee Atomic LOCA Group was responsible for addressing the LOCA aspects of this situation.[28]

We examined the use of the Best Estimate evaluation model to respond to those requests and the appropriateness of the response. In early 1990, due to difficulties in applying the generic Appendix K RELAP5YA SBLOCA code to the Maine Yankee plant within the prescribed time, the Manager had recommended a parallel alternative

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28 Id.

Best Estimate Evaluation Model [29] Since the previously submitted topical report did not include the Best Estimate approach, separate NRC approval would be necessary prior to implementation. The Manager recommended submittal of the Best Estimate model to the NRC, presumably for NRC review and approval. The Best Estimate approach was memorialized in a memorandum dated August 1, 1990. No NRC approval for the Best Estimate approach was sought by Maine Yankee. For some time, the Manager was under the impression that the Best Estimate approach methodology report had been submitted to the NRC by Maine Yankee for review.[30]

LOCA-91-04, dated January 25, 1991, and NED 91-18, dated January 28, 1991, which incorporates LOCA-91-04, are the memoranda responses to the 1990 service requests. In order to be responsive to the customer's request, Yankee Atomic used the only available LOCA analysis tool at that time, the Best Estimate model, to assess the effect of reduced steam generator pressure on the analysis of record. The author's characterization in LOCA-91-04 that the Best Estimate model was the licensing basis for SBLOCA analysis at the facility was incorrect and never corrected until the issuance of TAG-MY-93-012. The analysis of record at this time was the 1977 Combustion Engineering analysis. As noted by the NRC, the Manager did not sign the 1991 memoranda.[31]

The Technical Review Team disagreed with the NRC's premise that only approved Appendix K evaluation models can be used in performing some scoping safety evaluations, including input to 50.59 evaluations. The Technical Review Team found that models such as Best Estimate models can be appropriately used provided that the application does not replace the analysis of record, and provided that the use of the analysis method is clearly stated and justified. It noted that if there is any doubt regarding the appropriateness of such an application, then the NRC should be consulted prior to implementing the results of the analysis. It found that Yankee Atomic's failure in this situation was in incorrectly characterizing the Best Estimate LOCA analysis as the licensing basis analysis, and furthermore not stating that the analysis used non-NRC approved methods and had restrictions on its use.[32]

The Technical Review Team noted that the effect of reduced steam generator pressure on the SBLOCA results for the magnitude of steam generator tube fouling and plugging that was being evaluated would not be expected to be significant. This is particularly

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29 January 2, 1990 Memorandum to P.L. Anderson.

30 Technical Review Report at IV.D.

31 The Manager informed us that he likely would have recognized that the characterization of the Best Estimate model in the memorandum was incorrect.

32 Id.

true given that the analyses of record showed the SBLOCA PCTs to be lower than the LBLOCA PCTs. The Technical Review Team found that an evaluation could have been justified without any SBLOCA analysis. An evaluation could also have been justified using the Best Estimate analysis methodology provided that sufficient qualification was included, and provided that the analysis of record was not replaced. Given this finding, it would be possible to conclude that no technical inadequacy existed, i.e., that a non-approved LOCA evaluation model could be appropriately utilized to support a 10CFR50.59 analysis. However, we complete our analysis based on the assumption that use of the Best Estimate model was not appropriate.

We understand that the author of LOCA-91-04 may have believed he was preparing a scoping document to answer a hypothetical question from Maine Yankee for which the Best Estimate approach was appropriate. We reviewed the Service Requests from Maine Yankee which resulted in LOCA 91-04 and NED 91-18. We believe that based upon the amount of time permitted to complete the evaluation, it was questionable whether anyone contemplated that other than a scoping analysis was requested. The Technical Review Team agreed that a scoping analysis was requested. It is recognized that the Service Requests speak to bounding of the safety analysis; however, at the time, there was not the present sensitivity to the term "safety analysis" or to the need to analyze design basis deviations. Because the Best Estimate model was yielding significantly higher peak clad temperatures than the Evaluation Model of record (while still below the acceptance criterion of 2200°F set forth in 10CFR50.46(b)(1)), it may have been felt that it was more conservative to use this approach to determine whether the change in steam generator pressure caused the PCT to increase. In the circumstances, we conclude that there was a reasonable basis for the use of a Best Estimate model at the time despite the error in its representation as the licensing basis analysis.

The second element involves a response to the April 9, 1992 Service Request No. M-92-42, which asked for a determination of the minimum steam generator pressure that can be supported by analysis. A target value of no greater than 743 psig actual pressure was indicated as desirable. A request for the provisions of the "uncertainties that should be used with the computer and Main Control Board indications of steam generator pressure" was also made. In response to the request to determine the minimum steam generator pressure that can be supported by the analysis, on May 4, 1992 Yankee Atomic stated that this portion of the request "will be completed after the

revised SBLOCA model is completed."[33] The response indicated that the second half of the request had already been completed, referencing a memorandum, TAG-MY-92-030.[34]

The May 29, 1992 memorandum, TAG-MY-92-035, referenced NED 91-18. It is noted that the SBLOCA discussion in this memorandum is a very small part of the technical content and there is no mention of the referenced analysis as using the Best Estimate model. TAG-MY-92-035 was referenced by Maine Yankee in a 50.59 evaluation dated April 12, 1993 as part of the justification for operation with reduced steam generator pressure. The analysis based on the Appendix K SBLOCA results, i.e., RELAP5YA, were forwarded to Maine Yankee also on April 12, 1993, along with a draft 50.59 evaluation for Maine Yankee's use. These results were not referenced by Maine Yankee until January 13, 1994, when the original Maine Yankee 50.59 evaluation was revised.

We do not believe that it was appropriate to utilize NED 91-18 which relied on a Best Estimate approach without identifying it as being a non-approved LOCA model inasmuch as at that time, i.e., by April 1992, the LOCA Group knew that the Best Estimate model had not been submitted to the NRC. However, the Technical Review Team did conclude that its use would have been permissible provided that the application does not replace the analysis of record and provided that the use of the analysis method is clearly stated and justified. This was not done. Furthermore, in 1992, Yankee Atomic should have recognized that the Maine Yankee request was for more than a scoping analysis. We believe that Yankee Atomic should have recognized that its memorandum, TAG-MY-92-035, could possibly be utilized by Maine Yankee for safety related analyses.

Our next task was to determine whether the inadequacies described above were the result of careless disregard. In the 1992 time frame, utilities were only beginning to recognize that changes in plant conditions could constitute a change, test or experiment which would bring into play the requirements of 10CFR50.59. Whereas the steam generator pressure was an input to various station analyses, it was not a controlled variable nor do we understand that it could be found in the Technical Specifications for the facility. At the time, it was known that the LOCA small break analysis was not sensitive to the value of the steam generator pressure. This understanding was confirmed by later analysis. Generic Letter 91-18, "Information to Licensees Regarding Two NRC

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33 This response was contained in the YNSD Response section of Service Request No. M-92-42. We understand that it was the Manager's intent from April 1992 forward that the Appendix K SBLOCA analysis be used in conjunction with the analysis of the lowered steam generator pressure issue. However, this intent does not appear to have been communicated to the individual at Maine Yankee performing the 50.59 review.

34 We understand that as to the SBLOCA, the technical bases of the two memoranda, TAG-MY-92-030 and TAG-MY-92-035 are similar; both reference NED 91-18.

Inspection Manual Sections on Resolution of Degraded and Nonconforming Conditions and on Operability," which was issued on November 7, 1991, increased nuclear utilities' sensitivity of the need to prepare a 50.59 analyses for "as-found" degraded conditions, but such appreciation does not appear to have worked its way to the LOCA Group until much later. We do not believe that the period of time between the issuance of the generic letter and the issuance of TAG-MY-92-035 was such that it was likely that the isolated Yankee Atomic LOCA Group would have appreciated the effect of Generic Letter 91-18 upon the Group's activities.

The most compelling piece of evidence that we obtained on this issue was an indication that the impetus for Maine Yankee preparing a 50.59 evaluation related to the lowered steam generator pressure issue, i.e., based upon the May 29, 1992 memorandum, came from its NRC Resident Inspector some months after Maine Yankee received the May 29, 1992 memorandum. This acts to confirm that Yankee Atomic may not have been unreasonable in not anticipating that its memo would be used in a 50.59 evaluation. If the requirement to prepare a 10CFR50.59 evaluation was not within the contemplation of the utility which was in a better position to understand its design basis, the expectations of the NRC and its own procedures to implement 10CFR50.59, this is a strong indication to us that even if Yankee Atomic should have anticipated the design basis implications of its May 29, 1992 memorandum, its failure did not amount to careless disregard.

Also instructive are Yankee Atomic's actions and statements in an April 12, 1993, memorandum, TAG-MY-93-012, on the same subject matter. This memorandum incorporated the lower steam generator pressure of 735 psig in the safety analysis, utilizing RELAP5YA in the Appendix K mode as promised in the Service Request. With regard to compliance with 10CFR50.59, that memorandum stated in a section entitled "Safety Evaluation":

> This memo is safety related. It provides Maine Yankee with a single method for monitoring the full power SG pressure and overall heat transfer coefficient (UA) to verify that these parameters are within the safety analysis envelope. Since this memo theoretically expands the operating space of the plant from what is actually reported in Appendix D of the FSAR, it is our opinion that incorporation of the information into Cycle 13 operation may require a 10CFR50.59 evaluation. Attachment C contains 50.59 evaluation information (per Maine Yankee Procedure Number 0-06-4) which can be used by Maine Yankee to complete their formal review. It is not intended to be a complete or formal 50.59 evaluation. (emphasis in original)[35]

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35 It is ironic that this memorandum was dated the same day as Maine Yankee's approval date for its 50.59 analysis related to the May 29, 1992 memorandum. We understand that Yankee Atomic was not aware of Maine Yankee's 50.59 analysis at the time it issued the

We believe that this language reflects Yankee Atomic's new appreciation of the changing attitude in the nuclear industry as to the circumstances for which a 50.59 analysis need be generated.[36]

Our evaluation of reckless disregard considered the following factors:

1. The Best Estimate model was the only small break evaluation model available at the time.

2. The Best Estimate approach was yielding PCTs higher than the Evaluation Model of record.

3. The Best Estimate model was incorrectly indicated as being the licensing basis evaluation model.

4. In 1991, there was considerable confusion for reasons beyond the control of the LOCA Group as to the status of the Best Estimate model.

5. LOCA-91-04 could have been considered a "scoping" evaluation.

6. The Manager should have known that affirmative NRC approval of the Best Estimate model was needed and in 1992 that it had not yet been received.

7. There is no evidence that the Manager was familiar with the contents of NED-91-18 and LOCA 91-04.

8. The Manager apparently approved the May 29, 1992 memorandum without reviewing the references, including NED 91-18 and LOCA 91-04.

9. Failing to review all references was an error, but understandable.

10. Yankee Atomic personnel should have recognized that the May 29, 1992 memorandum had the potential to be utilized in a manner that affected the safety analysis.

11. From April 1992 forward, the Manager intended that the Appendix K SBLOCA analysis would be used to reanalyze a lower steam generator pressure limit than had

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memorandum.

36 Some time before this memorandum was prepared, Maine Yankee requested that Yankee Atomic prepare draft 50.59 input for any analyses it performed which could affect the facility's safety evaluation. We understand that the April 12, 1993 memorandum, TAG-MY-93-012, was among the first to incorporate such an evaluation.

been previously assumed, but such intent was not communicated to the individual performing the 50.59 review.

12. Maine Yankee personnel also did not contemplate the need for a 50.59 analysis relating to reduced steam generator pressure until suggested by the NRC Resident Inspector some months subsequently.

13. The regulatory position that degraded conditions would have to be reflected in the safety analysis via a 50.59 evaluation in the same way as affirmative changes in the design had only recently been addressed by the NRC. Apparently this guidance had not had an opportunity to have affected the thinking and actions of LOCA Group personnel with regard to steam generator pressure degradation.

14. Yankee Atomic's April 12, 1993 memorandum, TAG-MY-93-012, does contain a draft 10CFR50.59 analysis.

15. The Technical Review Team concluded that analytical techniques other than the approved Evaluation Model can be used to prepare 10CFR50.59 evaluations if sufficient disclosure is made.

We cannot conclude that there existed a reckless disregard or callous indifference toward their responsibilities or the consequence of their actions on the part of Yankee Atomic personnel. In the 1991-92 time frame, Yankee Atomic did not appreciate that steam generator pressure degradation affected the design basis; however, as this issue matured, so did Yankee Atomic's thinking as evidenced in its April 12, 1993 memorandum, TAG-MY-93-012. Yankee Atomic did not reasonably have knowledge that its behavior was likely to result in a violation of NRC requirements as it reasonably could have been expected to understand them at the time. Therefore, we conclude that there has not been careless disregard of the regulations.

VII. CONCLUSION

Section V.B of the Demand states as follows:

> An explanation why the NRC should not consider the inadequate analyses, which apparently caused MYAPCo to be in violation of NRC requirements, to be the result of willfulness, either deliberateness or careless disregard, on the part of YAEC and/or DE&S personnel.

The previous sections state the conclusions of the Willfulness Review Team as to its findings as to the specific matters within the scope of the Demand. The Demand focused on the actions of two individuals in relation to whether willfulness, either deliberateness or careless disregard, existed. We found no willfulness on the part of the two individuals or any other Yankee Atomic

personnel. We therefore believe that the Manager and Lead Engineer are capable of conducting their activities in the future in conformance with NRC requirements.

We conclude that there was nothing developed as a result of our investigation on the conduct of Yankee Atomic and/or DE&S personnel that would prevent the improvements that we understand are being made from being successful and resulting in DE&S' activities being conducted in full compliance with NRC requirements and expectations.

APPENDIX E

ROOT CAUSE AND CORRECTIVE ACTION ASSESSMENT

Report of a

ROOT CAUSE ASSESSMENT REVIEW

Performed for

DUKE ENGINEERING & SERVICES, INC.

Bolton, Massachusetts

Performed by Powerdyne Corporation

Reviewer/Preparer Donald C. Pruette Date 2/18/98

ROOT CAUSE ASSESSMENT REVIEW REPORT
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1.0 PURPOSE

On December 19, 1997, the U.S. Nuclear Regulatory Commission (NRC) issued a Demand for Information Letter (Demand) to Yankee Atomic Electric Company (YAEC) and Duke Engineering & Services Company (DE&S) to obtain information the NRC considered necessary to determine whether the addressees should continue to provide engineering analyses, and in particular Loss of Coolant Accident (LOCA) analyses, to NRC power reactor licensees. This letter was issued as a result of an NRC inspection finding that LOCA analyses performed by YAEC for Maine Yankee Atomic Power Co. (MYAPCo) were inadequate and that inaccurate information concerning those analyses had been provided to the NRC.

This report documents an independent Root Cause Assessment Review (RCAR or Review) performed by Powerdyne Corporation (Reviewer) of the discrepancies identified in the NRC's Demand. The purposes of this review were to:

(1) Validate that the root and contributing causes of activities described in the Demand letter had been appropriately identified;

(2) Validate that the corrective actions taken to remedy the identified causes were appropriate.

2.0 SCOPE

The scope of this RCAR included review of relevant activities regarding a previously performed YAEC Self-Assessment (performed in April 1996) and corrective action identification (a December, 1996 response to the Self-Assessment), reviews of documents associated with the discrepancies identified, review of relevant YAEC administrative and technical procedures, interviews with YAEC managers and other employees, and reviews of samples of analyses recently performed by the DE&S Bolton, Massachusetts Office. The Review was aimed at determining the validity of the identification of the root causes and corrective actions taken at the time of the Self-Assessment and their validity and relevance in the context of the current DE&S acquisition of the engineering organization of YAEC.

3.0 REVIEW PROCESS

The Root Cause Assessment Review was conducted during the period January 19, 1998 through January 23, 1998 in the Bolton, Massachusetts office of DE&S.

The process began with reviews of the NRC's Demand letter and a December 19, 1997 NRC letter to MYAPCo describing apparent violations stemming from discrepancies in their LOCA analyses which had been performed by YAEC in order to gain a clear understanding of the NRC's concerns, their understanding of the analyses discrepancies, and their position with regard to DE&S. Next, a report of the April, 1996 YAEC Self-Assessment was reviewed along with the December, 1996 Self-Assessment response by the engineering organization. This Self-Assessment had been prompted by the NRC's initial identification of the MYAPCo. LOCA analyses discrepancies. Next, two parallel assessment paths were pursued, reviewing relevant administrative and technical procedures and interviewing management and production personnel

in the DE&S organization who were responsible for the specific LOCA analyses that had been questioned by the NRC as well as individuals responsible for various aspects of other engineering analyses. And finally, samples of recent analyses were reviewed to determine if the types of concerns identified by the NRC or other concerns still existed and if the corrective actions had been effective.

The RCAR was conducted by Donald C. Prevatte, an engineering and management consultant with Powerdyne Corporation, a company providing services to the USNRC, the U. S. Department of Energy, and the nuclear power industry since 1982 (Appendix C provides details of the reviewer's qualifications).

4.0 BACKGROUND

4.1 History of the Concerns

In December 1995, the NRC conducted a technical review of the YAEC headquarters office in Bolton, Massachusetts and performed an investigation in response to allegations concerning the Maine Yankee LOCA analyses, which had been performed by YAEC. The NRC concluded from these activities that:

(1) By YAEC's preparation and approval of the RELAP5YA small break LOCA analysis (SBLOCA)and the WREM large break LOCA analysis (LBLOCA), and by YAEC's preparation and approval of the Core Performance Analysis Reports (CPARs) used to support Cycle 14 and Cycle 15 operation at Maine Yankee, YAEC caused MYAPCo to be in apparent violation of 10CFR § 50.46(a)(1).

(2) YAEC provided MYAPCo with information that was not complete and accurate in all material respects regarding this noncompliance with the above cited regulation, and thus caused MYAPCo to apparently violate 10CFR § 50.9(a) by maintaining CPARs which contained information which was not complete and accurate in all material respects in connection with MYAPCo's Cycle 14 and Cycle 15 reload analyses.

(3) As a result of its incorrect calculation of penetration factors, misapplication of the Lob-Chambre correlation, and inadequate review of YAEC-1868, YAEC caused MYAPCo to rely on an unacceptable evaluation model which overpredicted core cooling and overstated the conservatism of the evaluation model for Cycle 14 and Cycle 15 in apparent violation of 10CFR § 50.46(a)(1).

(4) By its use of an unacceptable "best estimate" SBLOCA analysis to determine the effects of a reduction in Steam generator pressure on LOCA analyses, YAEC caused MYAPCo to apparently violate 10CFR § 50.46(a)(1).

YAEC in April, 1996 initiated a Self-Assessment conducted by its Quality Assurance Department and documented in *RELAP5YA Self Assessment, Maine Yankee and Yankee Atomic*, 4/96, a report attached to YAEC Memorandum 632.TGS dated 4/30/96. This

effort identified the following three major common issues (root causes) in addition to numerous attendant and contributing causes for the NRC identified discrepancies:

(1) The divisions of responsibilities/ownership and roles of organizations performing, controlling, administering, and managing activities for the Maine Yankee plant were not completely and clearly defined or understood by all parties involved in or impacted by the activities.

(2) YAEC personnel lacked formal company training and retraining in areas that were essential to performance of their day-to-day tasks.

(3) Procedures for controlling analyses were weak in defining important processes, and these processes were driven by personal knowledge rather than by procedural guidance. Most specific to the NRC's concerns, the procedures did not require identification of the effects of analyses on licensing commitments or design basis documents.

Subsequent to the Self-Assessment, corrective actions were taken in the YAEC's engineering organization, and in December, 1996 a response to the Self-Assessment, Yankee Atomic Memorandum NEDMY96-052, 12/31/96, "Response to RELAP5YA Self-Assessment", was generated outlining those corrective actions.

On December 19, 1997, the NRC issued its Demand for Information Letter to YAEC and DE&S, to be responded to within 30 days (later extended to 60 days), and in January, 1998, DE&S initiated this independent RCAR.

4.2 Brief Company History - the Context for the Concerns

To clearly understand the concerns addressed by this Review and their root causes, one must understand their historical context, as was gleaned through interviews with several managers, including the YAEC Chairman and Chief Executive Officer who presided over the selling of YAEC's engineering services division to DE&S.

YAEC was one of the pioneers in commercial nuclear power in the United States, and was at the center of most of the commercial nuclear power endeavors in New England from the early years up until quite recently. New England was, and still is, somewhat unique, in that most of the nuclear power plants were owned by numerous, relatively small local utilities, many of whom held small interests in several of these plants. Few of these owners were large enough or sufficiently qualified to shoulder the complete responsibility of such a facility. YAEC, on the other hand, had the organizational entity with the size and qualifications needed to design, operate, and maintain these plant, and in the cases of the Main Yankee, Vermont Yankee, Yankee Rowe plants, YAEC actually held the original operating licenses. As a result of this multi-ownership, multi-plant situation, the relationships between the various utility owners and YAEC was, from the beginning, very familial, with the various roles and responsibilities understood by the parties mostly through their common historical perspectives rather than through explicit procedural

requirements, although, as with other nuclear organizations, procedural controls tended to increase with time.

In 1981, YAEC's relationship with Maine Yankee changed dramatically, with Maine Yankee taking over as the plant operating licensee holder. With this change, YAEC was no longer the primary point of contact for licensing related communications with the NRC for Maine Yankee. Although YAEC remained the primary engineering services provider, Maine Yankee could and did acquire services from other engineering firms. The precise organizational relationships and responsibilities between MY and YAEC, however, were not redefined or documented at this point, but facets of their former relationships appeared to have been assumed to remain by one or both parties, sometimes incorrectly. This was the environment in which the discrepancies discovered by the NRC were bred.

Another important historical context for these concerns was the relationship between YAEC and the NRC with respect to the reporting of LOCA analyses and the unique regulatory requirements applicable to LOCA analyses. Traditionally, these analyses were performed by nuclear Steam supply system (NSSS) vendors, such as Westinghouse or General Electric, who worked in close concert with the NRC in the development and application of the codes used. YAEC had not participated in this early development work, and therefore, was not familiar with the informal precedents and expectations that had evolved, although it was aware of the CFR-mandated formal reporting requirements for LOCA analyses. It was not aware, for instance, of the NRC's expectations for regular dialogue during the development stages of codes that had become standard procedure with the NSSS vendors during the early days of the business. During the application of the NRC-approved RELAP5YA code for the Maine Yankee SBLOCA, YAEC had concluded that further interactions with the NRC were not required based on a closeout letter which Maine Yankee received from the NRC. This unfamiliarity with the NRC's expectations concerning dialogue on these code applications, though not mandated by the CFR's, appeared to have been a contributing cause to the NRC's displeasure with YAEC's performance.

During the twenty-one month period from the Self-Assessment until this Review, two other historical events transpired that significantly affected the organizational relationship between YAEC and the NRC, and YAEC's present and future relationship with the NRC: (1) In August, 1997, the owners of Maine Yankee made the decision to permanently shut down and decommission the plant and (2) on December 1997, YAEC was acquired by DE&S. These events rendered many of the specifics of the Self-Assessment observations and responses obsolete. However, all of the principles embodied in the Self-Assessment report were still valid and applicable to the current situation. Therefore, this review addressed the findings and responses to the Self-Assessment in the context of the situation as it existed at that time, as well as the embodied principles as they applied to the current context - an established organization with longstanding proven nuclear power industry capabilities, in the midst of redefining itself - its organization and culture, its new roles and responsibilities, its new participants, and its significantly different business environment.

5.0 GENERAL OBSERVATIONS

This RCAR addressed each of the three major topics of concern identified in the Self-Assessment as well as other areas not addressed. The observations contained herein were based on a combination of reviews of the assessment and response documents (see Appendix B for a list of the documents reviewed), interviews with key managers in the former YAEC organization and with personnel with direct responsibilities for either production work performance or for support functions to the organization, such as training, quality assurance, and licensing (see Appendix A for a list of the personnel contacted), reviews of administrative and technical procedures (Appendix B), and reviews of a sampling of the production work (Appendix B). The following sections describe briefly the reviewers observations:

5.1 Perspectives on the NRC's Concerns

Two YAEC engineers were identified in the NRC Demand letter as being primarily responsible for the concerns with the LOCA analyses for Maine Yankee - one of the individuals who produced the analyses and the responsible Group Manager. Both individuals were interviewed in this Review to assure that their perspectives of the NRC concerns were captured. Summarized, the NRC concerns were:

(1) YAEC performed no specific code analysis demonstrating compliance with 10CFR50.46 in the LOCA break range from 0.35 ft^2 to 0.6 ft^2, the gap between the analyses capabilities of YAEC's small break and large break LOCA codes.

(2) YAEC did not adequately apprise MYAPCo of this analysis gap, thus causing them to violate 10CFR50.46 and 10CFR50.9,

(3) YAEC's 10CFR50.46 analyses used incorrect or misapplied factors and correlations, and because of inadequate QA, failed to detect these errors, causing MYAPCo to violate 10CFR50.46, and

(4) YAEC used an unacceptable "best estimate" small break analysis to resolve a concern with reduced Steam generator pressure, thereby causing MYAPCo to violate 10CFR50.46.

Both individuals contended that although, in hindsight, analysis errors and reporting errors were made, the NRC letter had incorrectly characterized these errors and had unjustly singled out YAEC and themselves as the cause of the concerns as follows:

(1) Although, as maintained by the NRC, there was no overlap in the range of break sizes for which specific code analyses were performed, the individuals felt that their judgement that the full range of the break spectrum was represented by the analyses that had been performed was reasonable, appropriate, justified, and valid at the time. This was based on several facts: (a) The trends of the results in the adjacent break ranges gave reasonable indication that peak cladding temperatures and the other critical parameters in the unanalyzed range would not peak or be higher than the results obtained in the analyzed regions. (b) They knew of no

contrary indications for code analyses performed for other plants of similar size and design or in their previous experience. (c) Two subsequent code analyses that addressed the unanalyzed range - one a formal analysis performed by Siemens and the other an informal YAEC analysis using the small break LOCA code with adjustments to render it more stable in this region - both confirmed the validity of their judgements, and further, showed that the YAEC analyses were, indeed, very conservative. (It was also noted by the Reviewer that 10CFR50.46 does not specifically require that the code analyses be overlapping for the complete spectrum of break sizes as seemed to be implied by the NRC's Demand letter, although it does appear to imply that where gaps exist, these *should be* brought to the NRC's attention.)

(2) The YAEC small break LOCA analysis report to Maine Yankee specifically identified code instability for break sizes above 0.35 ft^2, and these individuals stated that this code characteristic was discussed on several occasions with Maine Yankee's cognizant personnel. Although the NRC Demand letter acknowledged the report statements, it stated that the report's language would not signify to someone without LOCA code expertise that the code was not valid for breaks larger than 0.35 ft^2. However, the responsible YAEC individuals felt that the licensee audience for this report had been sufficiently qualified to understand the report's language, particularly in view of the attendant discussions that had taken place, and that therefore it was not reasonable to hold YAEC and themselves totally responsible for this miscommunication

(3) The YAEC individuals acknowledged the analyses errors that were identified by the NRC and the failure to identify them in the checking process, but they maintained that the effects of these errors on the results were insignificant.

(4) The YAEC individuals acknowledged that, in retrospect, using the "best-estimate" small break LOCA analysis to resolve the concern with the lower Steam generator pressure was an error, but pointed out that it was used with the full knowledge, understanding, and concurrence of YAEC management and the licensee's cognizant engineers.

Since the NRC's inspection that had initially identified these concerns, a great deal of these individuals' efforts had been directed toward defending against the allegations surrounding them. This appeared to have taken its toll on their ability to contribute to normal YAEC working endeavors and in their personal lives. Although the reviewer did not have the time to verify the accuracy of, or delve deeply into, the details of their individual stories, the perception derived from their interviews, other interviews, and other information gathered was that the personal damage these individuals had received and were continuing to receive was not commensurate with any technical or judgement mistakes they may have made.

5.2 Organizational Responsibilities/Roles/Communications

The first major common issue identified by the RELAP5YA Self-Assessment was that "The division of responsibilities/ownership and roles of organizations performing, controlling, administering, and managing activities for the Maine Yankee plant are not always completely and clearly defined or understood by all parties involved in or impacted by the activities." Many of the interviewees contacted in the Self-Assessment also felt that the relationship with Maine Yankee was changeable, from contractor to integral part of the Maine Yankee organization, depending on who was contacted and/or which relationship was most advantageous to Maine Yankee. The lines of communications/ reporting between the organizations and with external organizations, such as the NRC, were not clearly defined or understood. The Team recommended generating interface documents clearly defining the relationships between YAEC and Maine Yankee and clearly communicating these relationships to their respective staffs.

The Reviewer agreed with this assessment: it appeared that lack of formal definition of responsibilities, roles, lines of communications, etc. had been a major contributor to the analysis and reporting errors that were the focus of the NRC's criticism. The Reviewer believed further, based on interviews that revealed the common history of the two companies, that one of the *root causes* of this poor definition was this common history where such responsibilities, roles, and communications had been almost solely within the purview of YAEC as the license holder, and that this had been clearly understood, though not clearly documented, by both organizations during that early period. And that with the transfer of the license to Maine Yankee, YAEC had retained some of the previous responsibilities, but there had never been conscious, concerted decisions made in either organization concerning responsibility splits, with clear, written communications generated describing these responsibility splits. And further, for the vast majority of tasks undertaken by YAEC, the residual unwritten understandings between the companies after the license transfer had adequately provided quality work, and that the case in question was one of the few where these informal understandings had proved inadequate to provide the required assurance.

The response to the Self-Assessment on this issue found that Maine Yankee and YAEC managers were working to define the respective roles and responsibilities of their organizations working together and that an engineering responsibility matrix had been generated, but that further work was necessary by both organizations to refine the matrix and define the responsibilities. The response appeared to conclude that, at that point in time (December, 1996), the corrective actions had not gone far enough to assuage the problem.

The Reviewer agreed with the response conclusion; however, subsequent to the response, these managers appeared to have made further progress in defining their organizational relationships, but with the advent of the Maine Yankee shutdown, this issue with respect to this licensee, became moot. However, the more global issue, assuring clear definition of responsibilities and lines of communication between YAEC and *all* clients, an issue that the Self-Assessment Team had not been focused on, still remained.

At the time of this Review, this larger issue had been recognized and discussed by all of the managers interviewed, and a work control process outline procedure had been drafted in the Safety Assessment Group (part of the Nuclear Engineering Department) to address this issue. This procedure appeared to be very comprehensive, touching on all of the major points of concern, such as documentation of requested work from a client, definition of deliverables, tracking of work, scope definition, plan and schedule, client approval, work performance, and delivery and closeout. The reviewer could identify only a few additional significant points that were probably applicable to this procedure and should be incorporated as follows:

(1) Definition of the project or 'zational structure for all projects, identification by name the people in the or .zation positions, and identification by name of their counterparts or principal c .tact points in the clients' organizations.

(2) Detailed definition of the resources that will be required to accomplish a project, integrated with the project schedule, showing when and how these resources will be consumed.

(3) Routine tracking of scope accomplishment, meeting of milestones, and resources consumption tracking to assure that projects remain under control, i.e., delivering products within agreed upon schedules and budgets.

(4) Clearly defining the Quality Assurance and Licensing roles in all projects, or if they have no roles, clearly stating in project definition documents that they have no roles or that they have the minimal role of performing cursory reviews of project deliverables to assure that no new roles have been identified as a result of normal project evolution.

(5) Assuring that all these and the other points in the draft procedure are incorporated into any project contractual documents with DE&S clients to assure that scopes, roles, responsibilities, communications, etc. are clearly understood by *all* parties from the beginning.

5.3 Formal Company Training

The second major common issue identified by the RELAP5YA Self-Assessment was that "Personal at Yankee Atomic lack formal training and retraining in areas that are essential to performance of their day-to-day tasks." The Self-Assessment also concluded that the training should include the relationship and hierarchy of pertinent codes, standards, and regulatory requirements, and company procedures. Although read-and-sign level was a useful and significant training tool, it was not adequate alone; classroom instruction was felt to be necessary also to assure uniform and complete understanding of the codes, procedures, regulations, and management expectations.

The Self-Assessment Team recommended the following:

(1) That a formal training program, including classroom training, be established providing initial and periodic training on codes, regulations, and procedures pertinent to employees' areas of work responsibilities.

(2) That the Licensing Section Head be filled with someone specifically knowledgeable and dedicated to licensing coordination, or that a Licensing Engineer be specifically assigned to coordinate licensing activities.

(3) That a process be established to develop, review, document, and agree upon technical, regulatory, and licensing expectations of all matters in Project Engineering, NED, EED, PSD, etc.

The Reviewer agreed with the Self-Assessment Team's finding that a significant contributor to the discrepancies discovered by the NRC was lack of formal, job-specific training for YAEC employees and with the Team's recommendations.

The response memorandum to the Self-Assessment found that the only formal training performed to date had been on the reporting requirements of 10CFR50.9, 10CFR50.59, 10CFR50.72, and 10CFR50.72. This included a 3-hour classroom overview of the NRC, 10CFR50.59, and other reporting requirements, and a 1-day training session by Winston and Strawn. The response memo also documented pending and planed upgrades of the Maine Yankee licensing staff, but no changes in the licensing area at YAEC were identified. The memo also documented that Self-Assessment Team Recommendation 3 above for establishing a project expectations process had been fulfilled by a procedure established in NED, although this Reviewer was unable to locate the described procedure.

The Reviewer concluded from the individual responses in the response memorandum and through interviews that although sporadic, uncoordinated activities in training had taken place, no credible, coordinated, formal training programs had been established or even mapped out at that time, that there had been no *significant* activities in the YAEC organization related to generic coordination of licensing activities and reviews (this will be addressed in more detail in a later observation), and that a coordinated expectations process had not been established. Therefore, the Reviewer concluded that YAEC's response to the training concern identified in the Self-Assessment had been inadequate, and that, at that time, training was still inadequate to correct the concern.

The next question was, had subsequent activities in this area been adequate to correct the concern, i.e., (1) had a credible formal training program for employees had been established, and (2) if it had, was it effective?

The findings in this area were mixed and variable and were based on interviews with key managers, individuals performing work in the organization, an interview with the current company training coordinator, review of training related procedures, and inspection of sample training records.

Training Procedures:

Two procedures related to training were found and reviewed, WE-003, Rev 13, dated 6/27/97, *Indoctrination of Personnel*, and WE-004, Rev 12, dated 6/27/97, *Training*. These procedures were part of the Engineering Manual and appeared to adequately call out the training requirements for personnel performing engineering work, although it was not clear why two procedures were required to convey these requirements.

Procedure WE-003 contained a checklist form, Form WE-003-1, to document personnel training. Although this form covered many of the appropriate codes, standards, and regulations needed to effectively perform engineering work, it appeared to have the following weaknesses in the requirements it conveyed:

(1) It did not include the internal company procedures designated the Nuclear Engineering Department Procedures or the Technical Administrative Guidelines.

(2) In its list of pertinent NRC rules and regulations, it did not include 10CFR20, 10CFR50.9, 10CFR50.46, 10CFR50.71, 10CFR50.72, 10CFR50.73, and 10CFR50, Appendix K, two of which contain reporting requirements specific to the particular concerns identified by the NRC, and all of which contain reporting requirements pertinent to many day-to-day tasks engineering tasks in firms such as DE&S.

(3) 10CFR50, Appendix A was not identified as a mandatory regulation for review. This seemed incongruous for an engineering firm in the nuclear business, since Appendix A contains the NRC's general design criteria (GDC's) for nuclear power plants. Additionally, this list should have also contained the 1967 proposed GDC's, since many older plants were licensed to these rules.

(4) The categories of training did not appear to include classroom training, or previous specific work-related experience.

Based on these reviews the Reviewer concluded that the procedures were adequate but needed improvement.

Training Records:

The Reviewer selected two individuals in the Thermal Hydraulics & Safety Group to verify their training records. Although line managers and supervisors had the prime responsibility for employee training, such records were required to be maintained as a QA record in RMS. The company Training Coordinator was requested to provide these individuals' records. Although the records showed that both individuals, during their long tenure with YAEC, had received extensive valuable training in technical and work-related subjects, the records showed very little training in codes, standards, regulations, or company procedures specifically relating to their work. Due to time constraints, the Reviewer was unable to determine if this observation was due to records not being adequately kept of other training received, such training not actually having been received,

or the records of such other training stored in a different location (acceptable practice as long as RMS holds the official record copy).

Interviews:

Interviews revealed three different perspectives on training; from the eyes of the managers, the working level engineers, and the company Training Coordinator.

All managers and working level persons interviewed were not fully satisfied with the state of training as it existed. Although a comprehensive training program had been designed, the managers were frustrated because they felt that insufficient resources were available to support an effective training program. Additionally, since many of their people had been regularly required to work extended hours in recent years in order to meet client commitments, little time was left for training.

The company Training Coordinator was new to the job and appeared quite willing, dedicated, and conscientious. However, the Reviewer was surprised to discover that this individual had not been informed of the Self-Assessment Team's conclusion that inadequate training was one of the major causes for the discrepancies the NRC had identified, and that the Team had recommended corrective actions in the area of training. The scope of this position's responsibilities appeared to have been very narrowly defined to include primarily general and administrative training, e.g., employee benefits, EEO, interpersonal negotiation, time management, and similar areas, but no training in codes, regulations, or procedures, and no maintenance of employee records in these areas.

Putting all of this together, the Reviewer concluded that although a modicum of training applicable to the NRC's concerns was being performed, overall, the training program was inadequate to provide the necessary assurance to the NRC that this cause for the weaknesses they had detected was being corrected.

5.4 Analysis Control Procedures

The third major common issue identified by the RELAP5YA Self-Assessment was that "Procedures used for controlling the development of analyses are weak in defining important processes that currently appear to be driven by personnel [sic] knowledge rather than by procedural guidance. The current procedures do not require identification of the effects of analyses on licensing commitments or design basis documents." The Self-Assessment went on to say that "Yankee Atomic needs to overhaul the procedure for controlling the preparation and issuance of analyses to assure that regulatory limitations and thresholds are considered upon completion of calculations; to assure that licensing design basis documents affected by analyses are changed when needed; to assure that the use of computer codes are within the SER bounds for which they were approved; to assure that measures are in place for the use, control and dissemination of unverified data; and to assure that procedures appropriately address the QA program requirements." The Self-Assessment also found that there was general confusion as to who was responsible for updating the FSAR and other Maine Yankee licensing documents.

The Self-Assessment Team recommended that:

(1) Maine Yankee and YAEC management need to identify which organization has responsibility for FSAR update and assign a specific cognizant engineer this responsibility.

(2) Maine Yankee and YAEC should revise procedures to assure that appropriate trigger mechanisms are in place to prompt FSAR updates when needed.

(3) Training in licensing and regulatory updating requirements should be instituted.

The Reviewer agreed with the Self-Assessment Team's determination that analyses control procedures were weak and with their recommendations in principle, although, as with other recommendations, the Reviewer felt that they were too narrowly focused on just the relationship between Maine Yankee and YAEC; they should have focused instead on the same type relationships with all of YAEC's clients.

The response memorandum to the Self-Assessment on this issue documented that YAEC procedures had been revised to establish or lower thresholds and to ensure coordination, and six specific NED procedures were issued to address issues identified, including 10CFR50.46 reporting, review of computer software documentation, safety and relief valve modeling (an EDR finding), FSAR and Technical Specification review and updating, 10CFR50.59 safety evaluations, distribution of information affecting licensing commitments, closeout of service/work requests, and calculation checklists.

The RCAR Reviewer performed a cursory sampling of procedures used by YAEC engineers in performing their work (insufficient time was available to perform a detailed review), with particular emphasis on those procedures addressing calculations, analyses, and licensing reporting (see documents reviewed list in Appendix B). The aims of this sampling were to (1) determine if the particular concerns of the Self-Assessment, analyses control and licensing reporting, were adequately covered, (2) to verify that the programs and processes required by regulation were properly addressed, and (3) to ascertain if the procedures were clear, concise, understandable, coordinated with each other, and they made sense.

In general, the procedures sampled satisfied the Reviewer's aims; procedures were in place, had been recently updated, and appeared to provide very good control of analyses, and assured that reporting required by regulations was properly performed; the programs and processes required by regulation were properly addressed, including the requirements for a 10CFR50, Appendix B Quality Assurance Program in procedure YOQAP-I-A, communication with federal regulatory agencies in TAG No. 1, 10CFR21 reporting in TAG No. 6 and WE-109, 10CFR50.9 reporting in TAG No.18, 10CFR50.46 reporting in NED Procedure No. 5, and reporting of changes in analytical methods in NED No. 8; and the procedures reviewed were generally clear, concise, understandable, and made sense.

Based on this sample review and comments from interviewees, the Reviewer concluded that the weakness-in-procedures concern identified in the Self-Assessment had been

adequately addressed at the time of the response memo of December, 1996; that the specific concerns of the Self-Assessment, analyses control and NRC reporting had been well addressed in procedures; and that the company procedures had been further improved in the time since that response.

5.5 Licensing and Quality Assurance Involvement

One of the areas touched upon by the Self-Assessment Team was the roles of Licensing and Quality Assurance in the production of analyses. Their observations and the resultant responses were made in the context of the question, which organization, Maine Yankee or YAEC, and which part of the organization was responsible for assuring that the regulatory requirements and licensing commitments were satisfied and analyses work was performed in a quality manner? Although the specifics of their observations and the responses are obsolete today, the basic principles still apply; all work done on nuclear power facilities must be produced, reviewed, approved, applied, utilized, reported upon, etc., in the context of the regulatory requirements and licensing commitments applicable to that facility and must be done in accordance with a quality assurance program that conforms with 10CFR50, Appendix B. DE&S, therefore, must, as a part of its role as the consulting expert to whom clients turn for guidance, become more proactive in assuring that for all work proposed and performed by the Bolton Office, the specific areas of licensing and quality assurance responsibility, e.g., reviewing the FSAR, reporting to the NRC, determining the licensing bases, performing audits of the work, are accurately and completely identified, and the division of these responsibilities between the client and DE&S is clearly delineated and documented.

In this vein then, the Reviewer interviewed the Manager, Regulatory and Industry Affairs. The purpose was to ascertain if the necessity for routine Licensing involvement in future work, if no more than to determine that no involvement is required in certain tasks, was yet recognized in the new evolving mission/role of the DE&S Bolton Office. The interview revealed that, historically, YAEC's licensing roles had been mostly limited to two areas: (1) Providing direct NRC interface for clients, who were essentially other members of the Yankee Atomic family, where YAEC possessed specific technical expertise that required direct NRC contact, such as 10CFR50.46 LOCA analyses, and (2) for newly emerging fields and issues, such as decommissioning, where YAEC acted as the licensing vanguard for clients. Involvement in other day-to-day tasks for clients had not typically required Licensing's involvement. No specific plans had been made to address the more generic day-to-day needs indicated by the Self-Assessment findings and the changing role of the organization.

The only interview conducted in the Quality Assurance Department was with the Senior Quality Assurance Engineer that had directed the Self-Assessment. It appeared that early QA involvement with future work was a well recognized need at his level, probably as a result of his Self-Assessment experience. He also indicated that QA activity in the organization had been quite high in recent years with numerous other assessments having been made. This should also be another positive factor in responding to the NRC's concerns.

5.6 Thinking Outside the Box, or Taking the Broader View

As a part of the RCAR Review, the Reviewer, in consultation with the Manager of the Thermal Hydraulics & Safety Group, chose two recently generated calculations to review in order to ascertain the quality of the current production, review, and approval process. Both were associated with spent fuel pool cooling (see the documents reviewed list in Appendix B). The first was a thermal-hydraulics analysis of Seabrook's new spent fuel storage racks, and the second was an analysis of passive cooling in the Maine Yankee spent fuel pool for the loss-of-normal-cooling condition.

In both cases, the authors were well experience and qualified; in both cases, the objectives, inputs, and assumptions for the calculations were clearly and logically stated; and in both cases, there appeared to be no computational error.

The Seabrook calculation also contained sensitivity runs which gave good indication of model and code validity. However, a possible cultural weakness was revealed by this calculation. The model assumed no lateral communication between individual storage rack channels. The Reviewer knew from experience that some rack designs have longitudinal openings at the corners between channels that would allow lateral flow communication between channels. The Reviewer asked if the Seabrook racks had this design; if they did, the model used might have been invalid. The author did not know; she had not reviewed the rack hardware drawings. With subsequent review of these drawings the author verified that the channels were completely enclosed with no lateral flow paths, thereby confirming that model assumption. However, this indicated a possible process weakness; actual hardware documents not always being consulted to verify that models truly represent the conditions intended to be modeled.

The Maine Yankee fuel pool cooling calculation's purpose was to determine the temperature in the spent fuel pool as a function of time with no active cooling. This task was part of an overall project to "support the desired operation of the SFP in the post-shutdown condition". The underlying apparent intent was to determine at what point in time was it no longer necessary to provide normal forced cooling without exceeding an acceptable fuel pool temperature in natural circulation conditions. The Reviewer found several errors in this calculation as follows:

(1) The calculation did not include radiant heat transfer from the pool surface, although it was considered by the author and judged insignificant. At the higher pool temperatures addressed (190°F, 200°F and 210°F), this would appear, on initial inspection, to have been conservative. However, it was not, since this heat load added to the building would tend to raise building air temperature, thereby reducing the convective and evaporative heat transfer rates from the pool surface (the predominate heat transfer mechanisms), thus increasing the heat absorbed by the pool water, which would increase the rate of temperature rise, and thus reducing the time to reach the temperature limit.

(2) The evaporation rate, and therefore the heat transfer by that mechanism, was based on test relative humidities taken with the building HVAC operating, removing both

the water vapor and heat released inside the building from the fuel pool surface. For the condition for which the analyses were required, the availability of the HVAC could not be assured. Without HVAC, the building temperature and relative humidity would rise substantially above the calculation values (the relative humidity ultimately to 100%), thereby greatly reducing the effectiveness of the primary heat transfer mechanisms that were assumed.

(3) The evaporation rate was based on fuel pool level loss measurements taken over several days without makeup. However, no determination was made of any fuel pool leakage during that period. Such leakage is not uncommon and is normally detected by pool liner channel leakoff drains. This could have caused the results to be non-conservative.

In summary, the calculation set out to answer the wrong question, and in so doing developed an incorrect model (the control volume was too limited). The correct question should have been, at what point in time could the decay heat be removed **from the stored spent fuel to the environment by way of the building by entirely passive mechanisms** without exceeding the design limits on the fuel pool or any of the other associated equipment? An attendant question should have been, what would be the required water makeup rate to assure that under these conditions the water would not drop below an unacceptable level, considering that one of the primary heat transfer mechanisms would still be evaporation?

These discrepancies were discovered independently by the Reviewer; however, the author revealed during discussions that he had already discovered the overall concern of a too limited control volume, Condition Report 98-0003, dated 1/28/98 had just been written, and the licensee had been notified. The CR Committee and the author initiated a 10CFR21 evaluation process regarding this error since these discoveries were also considered to have potential implications for similar analyses performed by or on behalf of other licensees on other plants.

The Reviewer considered that in the context of one of the NRC's primary concerns, quality and control of analyses, these findings merit increased management attention. Discussions with the calculation authors and others led the Reviewer to conclude that a culture may exist in the Bolton Office, as in many organizations, of not taking a broad enough view of work assignments, not having a questioning attitude, not thinking outside the box; "the box" being the narrow boundaries thought to define one's working niche, in this last case, just performing an analysis but not defining the *real* questions, and in the first case, not consulting the design documents. In the last case, had the engineer taken the broader view of asking more questions of the licensee, helping the licensee define the *real* questions, these errors might have been avoided.

Another element of thinking outside the box is realizing that in all cases, the reason a client comes to experts such as DE&S is because he or she has a problem, but in many if not most cases, he or she does not have a total, clear understanding of the problem. Part of the expert's role is to help the client define and package the problem, figure out what the *real* questions are. Only by so doing can the consultant provide the true quality service

products that the client, company management, the NRC, *and* the people performing the work all truly desire.

6.0 SUMMARY CONCLUSIONS AND RECOMMENDATIONS

6.1 Regarding Perspectives on the NRC's Concerns

Conclusions:

(1) Although the DE&S individuals specifically identified by the NRC as having been primarily responsible for the analyses and reporting errors had indeed made some errors, the significance and mal-intent ascribed to these errors by the NRC demand letter seemed disproportionate to the circumstances as they appeared to the Reviewer.

Recommendations:

None.

6.2 Regarding Project Organizational Responsibilities/Roles/Communications

Conclusions:

(1) The Self-Assessment correctly concluded that a significant cause for the discrepancies with the Maine Yankee LOCA analyses and communications with the NRC was poorly defined organizational responsibilities/roles/communications between YAEC and Maine Yankee.

(2) The Self-Assessment was not focused on the more global underlying concern, that the company had no written requirements that such responsibility/role/communications definitions be clearly defined with *all* clients, not just Maine Yankee.

(3) At the time of the Self-Assessment response, although Maine Yankee and YAEC management discussions and plans had occurred concerning roles and responsibilities, and a responsibility matrix had been generated, little implementation of these plants had been effected.

(4) At the time of this Review, it was recognized by DE&S management that project definition was essential to assuring the quality of the company's work and the control of projects, and hence the company's continued success and profitability.

(5) The draft procedure, *NED Work Control Process Outline*, was a very good step in the right direction of establishing work control processes to assure the NRC and DE&S management that the quality of DE&S work for nuclear clients and other clients will meet their high standards and expectations.

Recommendations:

(1) That draft procedure *NED Work Control Process Outline* be completed and approved at the earliest practicable date.

(2) That the additional points outlined in Section 5.2 of this report be incorporated into this procedure.

(3) That references in this procedure to specific clients, e.g., Vermont Yankee and North Atlantic, be removed and that it be made generic and applicable to all clients.

(4) That this procedure be made applicable to all DE&S work performed for clients by the Bolton, Massachusetts office, not just nuclear engineering work.

(5) That procedures governing the day-to-day work of the "Contracts Group" in the DE&S organization be generated or revised to incorporate the applicable principles and provisions of this draft procedure.

6.3 Regarding Formal Company Training

Conclusions:

(1) Lack of formal employee training in the area of day-to-day work requirements, such as applicable codes, standards, regulations, and procedures was correctly identified by the Self-Assessment Team as a major contributing cause for the discrepancies with the Maine Yankee LOCA analyses and communications with the NRC.

(2) At the time of the response memo to the Self-Assessment Team's findings, very little had been done to develop and implement an effective company training program that would correct the problem.

(3) At the time of this Review, little additional work had been done to develop and implement an effective training program that would provide the desired assurance to company management and the NRC that this problem area had been corrected.

Recommendations:

(1) That a senior level management commitment of will and resources be made to develop and implement an in-depth company training program for all personnel providing services to nuclear utility clients with the goal of providing all such personnel with the specialized knowledge they require to perform their work in the quality manner expected by management and the NRC. (Although such a program would initially entail overhead costs, much of these costs could probably be recovered directly by offering elements of this training to clients, and indirectly by the enhanced credibility it would afford DE&S in the nuclear services market.)

(2) That a well-qualified individual be appointed as Training Manager, or similar title, and charged with the primary responsibility of developing and implementing the company's technical training program, and that this person be provided the authority and resources required to effectively carry out this responsibility.

6.4 Regarding Company Procedures

Conclusions:

(1) Weak procedures for controlling analyses and reporting of their effects on licensing commitments was correctly identified by the Self-Assessment Team as a significant contributing cause to the discrepancies found by the NRC in the YAEC LOCA analyses for Maine Yankee and for their concerns with reporting of LOCA analyse results and analyses changes.

(2) The weakness-in-procedures concern had been adequately addressed by the time of the response memo of December, 1996, and the specific concerns of the Self-Assessment, analyses control and NRC reporting, had been well addressed by company procedures.

(3) Company procedures reviewed appeared to have been continuously improved in the time since the response to the Self-Assessment memo of December, 1996.

(4) Some of the procedures still contain references to particular clients (vestiges of YAEC's previously close familial relationship with many of the New England plants), and virtually all of them refer to the company as YAEC. These appear inappropriate in the context YAEC's new evolving role as a technical services provider to more than just local clients, the acquisition by DE&S, and the acceleration of the evolution of the company's business as DE&S.

(5) Procedures do not appear to be optimally controlled, i.e., by a single company control procedure that spells out their format, authority, use, hierarchy, revision control, etc. As a result, formats are not consistent from one type to another, different types of procedures appear to address the same subject but contain different or complimentary or supplementary directions, and it is not always clear which procedural provisions are must-do requirements and which are simply recommendations. An example is several of the NED procedures which appear to address the same subject as the Engineering Manual procedures; these might be better controlled and present less potential for misunderstanding if they were combined into one procedure. Another example is the TAGs (Technical Administrative Guidelines) which contain directions concerning company policies and implementation of government regulations. The term "guidelines" appears to be misapplied for these procedures; they should be mandatory, must-comply procedures and should be identified as such in themselves and in a procedure-control procedure.

(6) The "Scope of Activities" section of the Engineering Manual introduction describes its applicability to design changes and additions but does not denote applicability to other engineering activities, such as performing engineering calculations, analyses, evaluations, etc., although it does contain specific procedures pertaining to these type activities.

(7) In NED Procedure No. 6, *Safety Analysis Process*, the term "safety analysis" is not defined, yet it appears to be describing calculations, for which there are two other procedures, NED No. 3 and WE-103. It is not clear from this procedure how a safety analysis differs from a calculation.

Recommendations:

(1) A complete procedure review and revision process should be initiated with all the company procedures to accomplish the following:

- Define the philosophy, policies, management expectations, and business goals that will represent the direction of DE&S's Bolton Office, and integrate these with the specific requirements from both the YAEC and the DE&S procedures into new DE&S procedures for the Bolton Office.

- Structure the new procedures to be generic with respect to clients, removing all specific references to YAEC's traditional clients, and the business practices and responsibility divisions that were unique to those relationships.

- Develop a procedure for procedures that spells out the format, authority, use, hierarchy, revision control, etc., for all DE&S Bolton Office procedures.

- Assure coordination and consistency between the requirements of the various procedure types and procedures in different organizations.

6.5 Regarding Licensing and Quality Assurance Involvement

Conclusions:

(1) As indicated by the Self-Assessment, the licensing and quality assurance facets of the YAEC organization had little involvement in the production, review, approval, delivery, and reporting of the Maine Yankee LOCA analyses that were the focus of the NRC's concerns, and that may have been a contributing factor to the discrepancies. Had these groups been involved as currently envisioned by the Reviewer, the potential for these discrepancies might have been somewhat reduced.

(2) The relative roles of Maine Yankee's and YAEC's Licensing and QA Groups was not clearly defined or understood by either of the organizations, as was concluded by the Self-Assessment.

(3) At the time of the response to the Self-Assessment, little had changed in YAEC to more clearly define those roles between these two companies or in the broader generic sense for YAEC.

(4) Even today it may not be fully recognized that Licensing and QA should be involved from the earliest stages of any work endeavor (the scope definition and proposal stages) to assure that, for all contracts, all of the licensing and QA tasks and responsibilities are identified and the assignment of these between the client and DE&S is clearly delineated and documented.

Recommendations:

(1) The definition of the roles and responsibilities of Licensing and Quality Assurance in the DE&S organization should be revised to include review, input, and client interface from the beginning of all contracts with nuclear clients, as well as other clients where licensing and QA functions may be required, in order to assure that all of the licensing and QA tasks and responsibilities are identified and the assignment of these between the client and DE&S is clearly delineated and documented.

(2) DE&S should establish and maintain a regular corporate presence in the NRC's regional and headquarters offices in order to keep them apprised of our capabilities and challenges in generic programs/analyses we are developing, to enable us to be more cognizant of generic industry concerns and NRC perspectives, and to provide assurance to the NRC that we are aggressively addressing their concern with poor communications between us and them.

6.6 Regarding Thinking Outside the Box, or Taking the Broader View

Conclusions:

(1) Based on a sample of two recently performed calculations and analysis related conversations, the Reviewer concluded that the quality and control of analyses may not be at the level desired; however, the small sample size was insufficient to base a final conclusion; further sampling would be required.

(2) A culture may exist in the Bolton Office of not taking the broader view, not having a questioning attitude, "not thinking outside the box" in the approach to work performance.

(3) A performance-based approach to reviews, i.e., sampling the product, in this case analyses, provides the ultimate indication of whether or not corrective actions have

been effective. In this case, initial samples indicated that corrective actions may not have been effective.

Recommendations:

(1) That a broader sampling of recent analyses be performed to provide a more accurate picture of whether samples by the Reviewer were anomalies or were true indications that corrective actions had not been effective. Subsequent corrective actions should be based on that sampling.

(2) The Reviewer believes that taking the broader view, developing a questioning attitude, and thinking outside the box is a learned skill; therefore, it can be taught. It is, therefore, recommended that a course of instruction be developed and provided to all Bolton Office employees on developing these skills.

(3) All analyses should be reviewed by a person or persons with very broad experience bases in the nuclear power industry. An essential element of this experience should include, but not be limited to, in-plant, hardware-oriented operational or startup experience, in order to assure that analyses reflect or envelop the reality of actual hardware and actual plant conditions.

(4) Where possible, actual hardware design documents should be used in the development of analyses models.

7.0 OVERALL CONCLUSIONS

The Reviewer concluded that the Self-Assessment identified most, but not all, of the root and contributing causes for the discrepancies identified by the NRC in their December, 1995 technical review. Most prominently missing from the Self-Assessment was identification of the apparent culture in the Bolton Office of not taking the broad view in the approach to work and responsibilities.

The Reviewer concluded that the corrective actions identified in the Self-Assessment were appropriate but that they had not all been effectively carried out. Most prominent was the inadequacy of company training to assure that employees have the knowledge of codes, standards, regulations, and procedures and other skills required to perform their job functions in quality manner.

The Reviewer concluded that problems may still exist in the area of quality of analyses; however, additional review should be performed in this area to determine if observations were anomalies or true indicators of a continuing problem area.

LIST OF PERSONS CONTACTED

Name	Title/Position/Function
Paul A. Bergeron	Manager, Thermal Hydraulics & Safety Group
Kathleen E. Bocon	Training Supervisor
James R. Chapman	Director, Nuclear Engineering Department
Don K. Davis	Chairman & Chief Executive Officer, YAEC
Cam DiNunzio	Senior Quality Assurance Engineer
Greg Hudson	Project Director
Dean A. Huggins	Manager, Records Management Services
Michael F. Kennedy	Manager, Safety Assessment Group
John M. Oddo	Manager, Regulatory and Industry Affairs
Suzanne Palmer	NED Engineer
Liliane Schor	Project Manager II
Stephen P. Schultz	General Manager, Nuclear and Fuel Services
Michael W. Scott	NED Engineer
Ramu K. Sundaram	NED Engineering Consultant

LIST OF DOCUMENTS REVIEWED

1. USNRC Independent Safety Assessment of Maine Yankee Atomic Power Company, Executive Summary, 10/7/96.

2. USNRC Letter to Mr. Charles D. Frizzle, President, Maine Yankee Atomic Power Company, 10/7/96.

3. RELAP5YA Self-Assessment, Maine Yankee and Yankee Atomic, 4/96.

4. Maine Yankee Letter to USNRC Chairman Shirley A. Jackson, CDF-96-192, 12/10/96, Independent Safety Assessment

5. Yankee Atomic Memorandum NEDMY96-052, 12/31/96, Response to RELAP5YA Self-Assessment.

6. USNRC Letter to Mr. Don K. Davis, President & Chief Executive Officer, Yankee Atomic Electric Company, and Mr. John F. Norris, President & Chief Executive Officer, Duke Engineering & Services Co., 12/19/97, Demand for Information to Yankee Atomic Electric Company and to Duke Engineering & Services - RE: Providing Inadequate Engineering Analyses and Materially Incomplete and Inaccurate Information to An NRC Licensee (NRC OI Report No. 1-95-050).

7. Engineering Instruction WE-002, Rev 13, 6/27/97, Design Document Control.

8. Engineering Instruction WE-003, Rev 13, 6/27/97, Indoctrination of Personnel.

9. Engineering Instruction WE-004, Rev 12, 6/27/97, Training.

10. Engineering Instruction WE-103, Rev 17, 9/19/97, Engineering Calculations and Analyses.

11. Engineering Instruction WE-109, Rev 4, 5/28/97, Engineering Deficiency Reports.

12. NRC letter to Mr. Michael B. Sellman, President, Maine Yankee Atomic Power Company, 12/19/97, Apparent Violations Stemming from NRC Office of Investigations Report Nos 1-95-050, 1-96-025, and 1-96-043.

13. Technical Administrative Guideline No. 1, Rev 12, 8/1/97, Communication with Federal Regulatory Agencies on Behalf of Yankee Clients.

14. Technical Administrative Guideline No. 6, Rev 25, 7/1/97, 10CFR, Part 21 Reporting.

15. Technical Administrative Guideline No. 18, Rev 1, 11/24/97, 10CFR50.9 Reporting Requirements.

16. Technical Administrative Guideline No. 22, Rev 2, 8/19/97, Self-Assessment Program.

17. Technical Administrative Guideline No. 25, Rev 1, 10/1/97, Condition Report System.

18. Nuclear Engineering Department Procedure No. 2, Rev 1, 7/14/97, Microfilming and Tracking of Approved Calculations.

19. Nuclear Engineering Department Procedure No. 3, Rev 0, 8/22/96, WE-103 Review Checklist.

20. Nuclear Engineering Department Procedure No. 5, Rev 0, 12/11/96, 10CFR50.46 Reporting.

21. Nuclear Engineering Department Procedure No. 6, Rev 2, 5/7/97, Safety Analysis Process.

22. Nuclear Engineering Department Procedure No. 8, Rev 1, 5/13/97, Changes to Analytical Methods.

23. Operational Quality Assurance Program, YOQAP-I-A, Rev 27, 12/20/96.

24. Calculation Number MYC-2005, Rev 0, 12/1/97, Spent Fuel Pool Passive Cooling.

25. Calculation Number SBC-835, Rev 0, 12/9/97, Spent Fuel Pool Thermal-Hydraulics Analysis - New Racks.

26. Service Request No. M-97-27A, 9/4/97, Post-Shutdown Safety Analysis.

27. Condition Report No. 98-0003, 1/28/98, Discrepancy with Maine Yankee fuel pool passive cooling calculation.

28. Draft NED Work Control Process Outline.

29. Excerpts from Maine Yankee Small Break LOCA Analysis, 6/93.

30. Maine Yankee letter to Mr. William T. Russell, Director, Office of Nuclear Reactor Regulation, USNRC, CDF-96-063, 4/25/96, Submittal of Maine Yankee SBLOCA Licensing Analysis in Compliance with 10 CFR 50.46 and in Satisfaction of TMI Action Items II.K.3.30, II.K.3.31, II.K.3.5.

31. Maine Yankee letter to Mr. Frank J. Miraglia, Acting Director, Office of Nuclear Reactor Regulation, USNRC, MN-96-145, CDF-96-180, 10/18/96, Response to USNRC Request for Information (RFI) - Maine Yankee SBLOCA Analysis.

32. Yankee Atomic - Bolton Memorandum, W. J. Metevia and R. K. Sundaram to P. L. Anderson, 1/2/90, Recommended Approach for SB LOCA II.K.3.31.

QUALIFICATIONS OF REVIEWER

Summary of Qualifications:

The Root Cause Assessment Review relating to the (date) NRC Demand-for-Information letter to Duke Engineering & Services, Inc. was performed by Mr. Donald C. Prevatte. Mr. Prevatte was well qualified to perform this review because of his extensive engineering and management experience in the nuclear power and other industries. This experienced spans more than 30 years in design, startup, testing, and inspection of nuclear and fossil power facilities, nuclear submarines, and jet engines, and management and direction of engineering organizations performing these activities. Approximately half of the Reviewer's experience is as an independent consultant providing engineering and management services to the power industry and government regulatory agencies, including the U.S. Nuclear Regulatory Commission and the U.S. Department of Energy, and practicing all of the skills necessary to operate a successful consulting business. The Reviewer has directed or participated in fifty-seven performance-based Team inspections of commercial and government nuclear facilities (see attached list). The Reviewer is a degreed mechanical engineer and a registered professional engineer. A more detailed description of the Reviewer's qualifications is provided in the attached resume.

POWERDYNE CORPORATION

7924 WOODSBLUFF RUN, FOGELSVILLE, PA 18051 (610) 398-9277, FAX (610) 398-0277

Resume of
DONALD C. PREVATTE

SUMMARY OF QUALIFICATIONS:

- More than 30 years of engineering and management experience in the design, startup, testing, and inspection of nuclear and fossil power facilities, submarines, and jet engines.
- Proven self-starter in an entrepreneurial environment, with a track record of running all aspects of a successful engineering consulting business.
- Highly analytical problem solving skills, with the ability to clearly define objectives, formulate logical, concise plans, maintain focus, and carry out plans.
- Outstanding organizational and time-management skills. Ability to prioritize, manage multiple tasks, meet schedules/deadlines. Attentive to both "big picture" and details.
- Capable people manager, with the ability to build Teams, delegate, match skills to jobs, provide technical oversight, define goals, generate motivation, recognize contributions, and deal with non-contributors.
- Innovative, persistent, dedicated, versatile, enthusiastic, and conscientious.
- Excellent written, verbal, and interpersonal communication skills.

EDUCATION:

Bachelor of Science Degree in Mechanical Engineering, North Carolina State University.

PROFESSIONAL EXPERIENCE:

1982 - present, Powerdyne Corporation

President, Independent Consultant - Providing engineering and management consulting services to the power industry. Responsibilities include P&L, marketing, proposal preparation, contract negotiation and preparation, nuclear and fossil plant systems and equipment design, engineering analysis, technical document generation, plant inspections, project planning and management.

1987 - Present: Participated in fifty-seven performance based Team inspections of nuclear power facilities for the Nuclear Regulatory Commission, the Department of Energy, and nuclear utilities. Performed reviews of design, maintenance, testing, operations, and human performance, and served as **Team Leader**.

1982 - 1992: Provided engineering and management consulting services to Pennsylvania Power & Light Company for the Susquehanna Steam Electric Station. Supervised the Engineering Planning and Scheduling Group, gaining experience in Critical Path Method project planning and controls using Project/2 computer code. Developed cost analyses and position papers to prepare client for a Public Utilities Commission management audit and to support a rate increase request. In the Power Uprate Project, performed system design evaluations, engineering analyses, and systems redesign to support a 5% power uprate. Evaluated the reactor building HVAC and chilled water systems, portions of the emergency service water system, the standby gas treatment system, the main condenser, and motor operated valves with regard to NRC Generic Letter 89-10 requirements. Performed analyses of reactor building heat loads and building coolers. In the Design Bases Document Project, performed licensing requirements/commitments research. During plant startup and early operations, generated analyses of high energy pipe breaks, two-phase jet impingement, and suppression pool heatup, and directed leak-before-break analyses. Developed plant modifications and procedure changes to mitigate MSIV leakage, and generated modifications for the diesel generator stating air system and the auxiliary boiler feed pump seal cooling system.

1987: Updated/redesigned the 10CFR50.59 Safety Evaluation Program for Point Beach Nuclear Plant.

1981 - 1982, General Physics Corporation

Director, Power Engineering - Directed twenty-eight engineers in design, analysis, and field engineering activities. Responsibilities included technical supervision, marketing, recruiting, planning and budget development. Directed Appendix J containment retests at the Brunswick Steam Electric Plant.

1980 - 1981, Quadrex Corporation

Site Manager - Supervised nineteen Quadrex engineers assigned to PP&L's Susquehanna Steam Electric Station, Nuclear Plant Engineering Department. Developed the conceptual design of the On-Site Low Level Radwaste Holding Facility and provided project management consultation.

1977 - 1980, Brown & Root, Inc.

Discipline Supervisor - Electrical and Instrumentation and Controls Design Groups on the South Texas Project. Responsible for the technical direction and production output of these disciplines. Developed a multi-discipline design review program. Supervised the rewrite of the System Descriptions.

Lead Mechanical Startup Engineer - Developed startup schedule, wrote test procedures, and performed "hands-on" testing at Parish Generating Station (fossil). Supervised boiler chemical cleaning/steam blows.

1972 -1977, United Engineers & Constructors, Inc.

Lead Startup Engineer - Successfully marketed the preoperational testing program contract for the Salem Generating Station, Unit 2. Developed the program and supervised sixteen startup engineers.

Senior Startup Engineer - At the Brunswick STEAM Electric Plant, responsible for the reactor vessel hydrostatic tests, all containment leakage rate and structural integrity tests, integrated system flushes,

diesel generator acceptance tests, various systems preoperational tests, and the main condenser tests. Coordinated systems completion for testing. Directed the first plant outage after full power operation.

1971 -1972, Newport News Shipbuilding & Dry Dock Company

Mechanical Test Engineer - Wrote procedures and performed post overhaul testing of submarine reactor plant systems. Attended Shift Test Engineer School.

1969 - 1971, Pratt & Whitney Aircraft Corporation

Design Engineer - Designed jet engine hardware for F-15/F-16 fighter aircraft. Performed finite element fatigue analyses using state-of-the-art computer codes and hardware.

REGISTRATION: Registered Professional Engineer, Texas.

MILITARY SERVICE: US Navy, Lieutenant, Navigator, Main Propulsion Assistant, Legal Officer.

TEAM INSPECTIONS PARTICIPATION BY D.C. PREVATTE

Plant	Client	Systems	Inspection Type/Area
Turkey Point	NRC	Intake Cooling Water	SWOPI Overview/Design
Sequoyah (2 inspections)	NRC	Emergency Raw Cooling Water Various	SWOPI/Design SALP Followup Inspection/Design
Hatch	NRC	Plant/RHR Service Water	SWOPI/Design
Farley	NRC	Emergency Service Water	SWOPI/Design
Arkansas Nuclear I	NRC	Emergency Service Water	Diagnostic Evaluation/Design
Zion (3 inspections)	NRC NRC NRC	Emergency Service Water Service Water/CCW Service Water	Diagnostic Evaluation/Design Material Condition Evaluation SWOPI Followup/Design
Watts Bar (2 inspections)	NRC NRC	Essential Raw Cooling Water Component Cooling Water	Integrated Design Inspection Vertical Slice Review Evaluation
San Onofre	NRC	Saltwater/Component Cooling	Integrated Testing/Design
Oconee	NRC	Emergency Service Water	SWOPI/Design/Maintenance/Testing
Surry	NRC	Emergency Service Water	SWOPI/Design/Maintenance/Testing
Palo Verde (2 inspections)	NRC NRC	Essential Service Water Emergency Power	Systematic Test Performance Diagnostic Evaluation/Design
Catawba	NRC	Nuclear Service Water	SWOPI/Design/Maintenance/Testing
Crystal River	NRC	Nuclear Service Water	SWOPI Overview/Followup/Design
Maine Yankee	NRC	Component Cooling Water	SSFI/Design
WNP-2 (2 inspections)	NRC NRC	Emergency Power Various	SSFI/Design SSOMI/Design
Cooper (2 inspections)	NRC NRC	Emergency Power SLC/CRD/RPS	SSFI/Design SSFI/Design
Robinson 2	NRC	Emergency Power	SSFI/Design
Fitzpatrick	NRC	Emergency Power	SSFI/Design/Maintenance
Rancho Seco (2 inspections)	NRC NRC	Emergency Power Auxiliary Feedwater	SSFI/Design SSFI/Design
Point Beach	Utility	Emergency Power	SSFI/Design
Diablo Canyon	Utility	Auxiliary Feedwater	SSFI/Design
McGuire	NRC	Auxiliary Feedwater	Diagnostic Evaluation/Design
Monticello	NRC	Core Spray	SSFI/Design

Plant	Client	Systems	Inspection Type/Area
9 Mile Point 1	NRC	Core Spray, HPCI	SSFI/Design
Trojan	Utility	Control Room Ventilation	SSFI/Maintenance
Grand Gulf	Utility	Standby Liquid Control	SSFI/Design
(2 inspections)	NRC	Standby Service Water	Engineering Team Inspection
Duane Arnold	Utility	HPCI	SSFI/Maintenance
Shoreham	Utility	Evaluation of SSFI/SSOMI	SSFI/Design
Browns Ferry	Utility	RHR	SSFI/Maintenance
(2 inspections)	NRC	Various	Modifications/Design
Wolf Creek	NRC	Various	SSOMI/Design
(2 inspections)	NRC	ESW, AFW and Diesel Gen.	Engineering & Technical Support
Dresden	NRC	Various	SSOMI Followup Inspection/Design
(2 inspections)	NRC	Various	Engineering & Technical Support Insp.
North Anna	NRC	Various	SSOMI/Design
Hope Creek	Utility	Liquid and Solid Radwaste	System Functional Evaluation
Salem	Utility	Various	Outage Modification Inspection
(4 inspections)	Utility	Radwaste Systems	SSFI/Design
	Utility	Fuel Handling Building HVAC	SSFI/Design Response Team
	Utility	Control Room HVAC	Design Review Inspection
Fermi II	NRC	HPCI, Inst. Air	SSFI/Design
Savannah River	DOE	Various	Human Performance Inspection
D. C. Cook	NRC	Centrifugal Charging	SOPI/Design
Kewaunee	NRC	AFW, Auxiliary Coolant	SOPI/Design
Ft. Calhoun	NRC	AFW, CCW, Raw Water	Engineering & Technical Support/Design
Millstone 3	NRC	Charging System	SSFI/Design
Paducah Gaseous Diffusion Plant	USEC	Autoclave System	SSFI/Team Leader/Design

APPENDIX F

ENGINEERING PROCESS ASSESSMENT

Assessment Report

of

Engineering and Technical Work Processes

Utilized at

DE&S' Bolton Office

Joe B. Stringer, Team Leader

2/23/98
Date

Assessment Report
of
Engineering and Technical Work Processes
Utilized at
DE&S' Bolton Office

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Assessment Report
of
Engineering and Technical Work Processes
Utilized at
DE&S' Bolton Office

1.0 Introduction

This report provides the results of an assessment conducted Feb. 2-6,1998 of the engineering and technical work processes utilized at the DE&S' Bolton office. The overall objective was to confirm the adequacy of management, work practices, work products and qualification of personnel performing work.

The assessment was performed by a DE&S team independent of the line organization at the Bolton location. The review team consisted of the Team Leader, a Quality Assurance (QA) expert, and three technical specialists, including an engineering manager consultant. The review team member assignments were made consistent with each member's area of expertise and experience. Team members and their areas of expertise are summarized in Appendix A.

An assessment plan, which was developed in accordance with the review team Charter, was approved Jan 30,1998 and used as the base document for the assessment. Criteria and review approach document worksheets were utilized to systematically conduct and document results of the assessment of selected work processes and associated focus areas. These worksheets provided overall and specific acceptance criteria for each topical area reviewed based on the performance issues stated in the review team charter. Specific acceptance criteria were utilized as a means for determining if the overall acceptance criteria had been met.

2.0 Purpose

DE&S' management initiative was to assess the adequacy of engineering and technical work processes, deliverables, and the competence of the personnel associated with these activities. In addition, the assessment evaluated compliance with accepted industry practices and regulatory requirements.

3.0 Scope

This assessment targeted engineering and technical support activities other than the loss of coolant accident (LOCA) analysis work which has been completed in the past three years. The review team used a combination of engineering and technical document reviews and personnel interviews to obtain information for the evaluation. After reviewing the current organization at the Bolton office the team identified several organizations to focus evaluations based on the work performed in each area, the time

available for the assessment and the focus of the evaluation (compliance with NRC requirements and safety related work). Organizations included:

- Nuclear and Fuel Services
- Decommissioning
- Automation Technology
- Environmental Health and Safety
- Quality Assurance

Work processes were assessed with a focus on performance of calculations, analyses, and computer code validations. Work processes included:

- Design and Design Changes
- Design Document Control
- Organization and People Deployment
- Training
- Performance Management

4.0 Results

The review team identified change over the past several years to be pervasive and as a result set the context for much of its interaction with personnel and review of work practices. The former Yankee Atomic Electric Company (YAEC) organization matured with well-defined, comprehensive responsibilities for the plants in its family. This organization maintained a strong project management function, performed the design engineering necessary to support the operation of these units, performed the licensing with the NRC and, at one time, held the operating licenses for these units. The interface with the clients was well understood and handled efficiently by the project management organization. Costs were addressed satisfactorily in the budgeting process. Work was performed in accordance with the YAEC procedures, and this organization was clearly responsible for its work and for maintenance of the design basis. In the execution of these responsibilities, the YAEC organization developed strong talents, a rich base of experience, and a culture of pride and ownership of its work, all of which remain evident today.

The current work environment, even before the acquisition by DE&S, is much different. In the case of the Yankee Rowe plant, the Bolton office still has relatively well defined responsibilities, including use of its own procedures with clear responsibility for approval of work products. However, in the case of Maine Yankee, the organization functions in an engineering support role. Maine Yankee defines the scope of the work, sets the schedule, approves the work, and is responsible for the interface of new work to the existing design and maintenance of the design basis. Work for other clients generally falls between the two extremes of Yankee Rowe and Maine Yankee.

As a result of these changes, work within the Bolton office is initiated by different methods, done to different procedures, subject to different QA requirements, and performed to different levels of responsibility for the work, all depending on the client. The pressure of scheduling was a common theme in our review throughout the organization. In the absence of the former project management organization, current methods of addressing schedule concerns both internally and externally are

immature. However, the commitment to work product quality in the face of schedule pressure was also a common theme.

The review team interviewed 32 managers and engineers, and reviewed approximately 60 documents, such as procedures, calculations, analyses, technical reports, drawings, memorandums, training records and audits. The list of individual sources of information is provided in Appendix B. Based on the assessment, the review team concluded that the acceptance criteria were satisfactorily met. Appendix C contains document worksheets that address the following assessment topics:

- Topic #1 - Engineering Process and Adherence to Applicable Procedures, and Industry and NRC Requirements
- Topic #2 - Design Controls, Including Engineering Documentation and Quality of Work Products
- Topic #3 - Management Control of Safety-Related Activities
- Topic #4 - Design Engineering Training; Particularly, QA Program Training
- Topic #5 - Management Control of Safety-Related Activities; Adherence to Applicable Procedures, and Industry and NRC Requirements; and Quality of Work Products

These document worksheets provide results of the assessment against specific review criteria and a conclusion at the end of each worksheet. Following is an overview of each organization evaluated, and conclusions and observations for each.

Nuclear and Fuel Services:

Fuel Management Services responsibilities include fuel materials procurement, fuel management analyses, and other related activities which are primarily commercial. This group has a focused work scope which has remained relatively unchanged during a period of other significant changes for the Bolton office. In general, the work of this organization is not safety-related, although it has an important oversight role. Any safety related calculations or analyses are performed in accordance with WE-103. A Procedure and Guidelines Manual provides an administrative framework for a number of its activities. As an example, it provides a systematic process for conducting and documenting the review of core component design changes. Work scope and work processes are well-defined and followed based on review of documentation. Engineering work products were found to be intact.

Nuclear Engineering responsibilities include reactor physics, core reload design, thermal hydraulics analysis, probability risk assessment and reliability analyses, fluid systems engineering and special projects. Approximately 80 percent of the work is safety-related, and WE-103 and WE-108 are the principal procedures applicable to their work. However, in response to previous audits, they have developed supplementary procedures to assist personnel in application of these WE procedures. As with other groups in the Bolton office, loss of the former process in which the project manager controlled work coming in has resulted in several methods for work initiation, including formal service requests, work orders, contracts, verbal requests and initiative for routine budgeted work.

The primary concern of the group is the schedule pressure associated with their work. Several techniques are employed to address control of in-house work and schedule concerns. They are negotiating with the plants to prioritize projects. They meet with the clients once a month to ensure control of work, and quarterly meetings are held with the plant reactor engineers mostly on core reload work. An internal

tracking system is used for all work and is reviewed weekly. Work is planned so that reviewers, independent verifiers and approvers are available to meet schedules. Memorandums are used to formally transmit information to the client. Scoping meetings are held with the client to define product and determine how the product will be used. Based on review of documentation overall in the Nuclear Engineering Department, safety-related work is performed in accordance with appropriate procedures, and industry and regulatory requirements.

Decommissioning:

Within the decommissioning organization there is one main engineering support group, Decommissioning Engineering & Services led by Jane Grant. This group was the focus of the assessment for the Decommissioning Department. Generally, the Decommissioning Engineering & Services Group provides engineering and licensing services for decommissioning of Yankee Rowe, Connecticut Yankee and Maine Yankee. Other projects are ongoing, but the focus of work at this time is the three listed plants/projects. The responsibilities from a management and project controls standpoint vary for each project.

Based on its past relationship with the Yankee Rowe plant, the DE&S Bolton group generally serves as the lead organization for engineering support work and project management. Working together with plant management, work tasks are identified and agreed upon by both parties. A Decommissioning Services Agreement between Yankee Atomic and DE&S provides contract terms and responsibilities for all work. A comprehensive procedure, AP-0103, provides details for completion of Decommissioning Work Packages (DWPs) and all associated tasks. The procedure covers the steps for initiating and completing work. Overall, work is defined in a service request and task description. A complete listing of current and completed tasks was provided for information.

For Connecticut Yankee, primary project responsibility is at the plant and tasks are completed utilizing a service request process, much like Yankee Rowe. Generally, work can be described as engineering support since most tasks are very specific in nature, compared to complete project responsibility at Yankee Rowe. Tasks are completed within a defined engineering support organization within the Decommissioning Engineering and Services group. Service requests are completed and signed initiating specific tasks. The service request also indicates if QA is needed. If so, then a Task QA Plan is completed, providing specific QA procedures and processes. Completion of work activities is clearly documented in memos to the client, Connecticut Yankee Atomic Power Company.

The work at Maine Yankee can be generally described as providing engineering support. Tasks such as WO# 00082.00.0009.01.00000 are described as Providing General Engineering for Decommissioning. The overall project responsibility is in a project organization lead by Entergy. We received a project organization dated Jan 20, 1998, which shows Bob Fraser of Entergy as the Director of Engineering. Work is performed within and under the direction of the project organization. This relationship and DE&S current work scope were clearly explained by Bruce Holmgren, Manager of Decommissioning Engineering, and Bill Henries, Project Engineer for the SNF Island DWP and lead mechanical engineer for Maine Yankee.

All work activities for the three projects focus on tasks associated with decommissioning the various systems, structures and components at the plants. Examples of ongoing engineering and design activities include design of the spent fuel pool island at Maine Yankee and various component removal activities at

Connecticut Yankee. Review of specific work packages and interviews indicated personnel understood the scope of work, what QA procedures apply and had a clear technical understanding of the work in the mechanical/structural, electrical/I&E and systems engineering disciplines.

In reviewing work scope, schedules and deliverables with various personnel, it became clear that there are multiple clients and needs. The available resources do not adequately meet the project needs, leading to challenges in maintaining quality. Personnel interviewed clearly understood the need for maintaining quality. In one instance, based on client insistence for completion of a deliverable on schedule, significant open items were identified within the document. When asked how the open items would be cleared, personnel explained a reasonable solution in accordance with existing procedures. Resource allocation among multiple clients seems to be creating some unnecessary challenges. Work products are sound and are being performed in accordance with appropriate procedures, and industry and regulatory requirements.

Automation Technology:

The Automation Technology Department provides support to the engineering groups on software development and revision, administers the software library, performs computer runs to test software, and provides electronic data management and computational services. Verification and validation are performed for each code, each revision and each platform. Work is performed under Procedure WE-108 when safety-related work is involved. A review of controlled computer codes was provided and based on selective review it appears all applicable procedures are followed.

Environmental Health and Safety:

The Environmental Health and Safety Department is responsible for radiation protection and health physics services at the site, environmental/radiological laboratory services, radiological engineering and environmental services. This review focused on the two latter organizations since their activities primarily involve the performance of design calculations and analyses. The Radiological Engineering Group performs radiological analysis in support of plant Final Safety Analysis Report (FSAR) maintenance, radiological and non-radiological assessments, meteorological dispersion analysis and radiation shielding analysis. The Environmental Services Group provides analysis in support of the Individual Plant Evaluations (IPE) extreme events analysis, floods, airline crashes and seismic events and potential consequences. They also provide non-radiological environmental services in support of permits and hazardous waste disposal. The coordination and business processes used to deal with budget and schedule type issues with the client have changed. In the past, these groups interfaced with the Projects Group at the Bolton office who in turn would interface with the client. Now this has changed and interfacing with the client is somewhat unclear. However, the technical interfaces with counterparts at the client locations have not changed. Work is initiated through multiple methods such as service requests, verbal requests that are followed up with service requests, and routine work. Results of their work are normally transmitted formally via memorandum. A significant number of analyses performed use computer codes. These are originated and maintained in conformance with the WE-108 QA procedure. Based on selected reviews of documentation and interviews with department personnel, work was found to be sound and in accordance with applicable procedures.

Quality Assurance:

The QA organization provides QA program requirements for the DE&S' Bolton office. They ensure QA requirements are integrated into work procedures and appropriately followed based on the conduct of audits. The role of QA in procedures and audits both within DE&S' Bolton office and with the sponsor plants, i.e., Maine Yankee, Vermont Yankee, Yankee Rowe, and Northeast Utilities (Millestone, Connecticut Yankee, and Seabrook) appears to be evolving. The QA function is currently transitioning to DE&S' QA Manual which is governed by NQA-1,1989 versus the Yankee QA program which is governed by ASME N45.2.11,1974. Some work is performed to the Bolton office procedures, others are performed by the procedures of the client. The mechanism by which QA requirements are defined for work tasks is handled differently depending on the client. Many of the Yankee sponsor plant procedures refer to the Bolton office procedure set. Interviews with QA personnel and review of QA documentation demonstrate that the QA department effectively supports the Bolton office in achieving its goals.

5.0 Conclusions

The overall conclusions of the review team are :

- Procedures are effective and correctly implemented.
- People are appropriately trained, qualified and knowledgeable.
- Management control of safety-related activities is evident.
- Engineering and technical work products are sound and in accordance with industry and regulatory requirements.

In addition, the review team identified a number of strengths. They are as follows:

- A strong culture exists that is conscious of safety and quality requirements, demonstrating clear evidence of pride and ownership of work.
- Personnel have in-depth technical knowledge.
- Client interface is strong and close.
- Initiatives are ongoing to improve the engineering work control process.

Three areas for potential improvements were identified:

A. DE&S Transition Plan

The review team discovered that a plan had not been finalized for transitioning from the Yankee organization to an approach that utilizes the best practices of Yankee and DE&S. Several activities were on hold awaiting issuance of the Transition Plan (e.g. enhanced training and self-assessments. as identified in Condition Reports). In addition, the plan needs to address existing contracts with clients and integration of QA program requirements, clear understanding of organizational and functional responsibilities, and administrative procedures and processes. The review team acknowledges that an effort is underway to develop and implement a transition plan which should address these issues.

B. Work Processes

A number of work process-related opportunities for improvement were identified during the assessment. It was found that different processes are used to initiate work and that the vehicle by which QA requirements are defined for work tasks is handled differently. It was also discovered that different levels of involvement in development of task schedules exist and that different methods for tracking work and progress are used. Unreasonable requests for resources were also perceived to have some impact on work process performance. It is recommended that the implementation of the project controls methodology be included in the Transition Plan.

C. Roles and Responsibilities

Several issues were found to exist involving manager and engineer roles and responsibilities. Roles and responsibilities are evolving and are currently different for each project/plant. Configuration management methodology is not clearly documented. It was not evident that there was a good understanding of roles and responsibilities at all levels of the organization. This appears to be a result of change and the organization's desire to comply with client demands.

6.0 Appendices

A. Review Team Membership
B. Information Sources
C. Completed Criteria and Review Approach Document (CRAD) Worksheets

Assessment Report
of
Engineering and Technical Work Processes
Utilized at
DE&S' Bolton Office

Review Team

Membership

Review Team Membership

Name	Company	Areas of Expertise
Team Leader		
Joe Stringer	DE&S - Charlotte Office	• Nuclear and hydro facility system design and startup • Equipment procurement • Project management • Engineering assessments • Radioactive waste management design and licensing
Team Members		
Robert Bonisolli	DE&S - Bolton Office	• QA program and procedures • Internal and external • audits/assessments • QC inspections • Training • Vendor qualification
Tom Wyke	Engineering Manager Consultant	• Nuclear facility and systems design and startup • Nuclear engineering and licensing support • Civil, environmental and electrical design • Nuclear relations with external organizations (e.g. NRC, INPO, WANO, EPRI, etc.)
Bob Eble	DE&S - Vienna Office	• Nuclear facility design • Nuclear analysis • QA procedures • NRC licensing criteria
Bob Futrell	DE&S - Charlotte Office	• Process reengineering • Nuclear safety & event analysis • Safety review and assessment • QA audits • Radiation protection • Operational readiness reviews and assessments

Assessment Report
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Information Sources

Persons Interviewed/Department:

1. Stephen Schultz/Nuclear and Fuel Services
2. James Chapman/Nuclear Engineering
3. Michael Kennedy/Safety Assessment
4. Paul Bergeron/Thermal Hydraulics & Safety
5. Fred Carpenito/ Thermal Hydraulics & Safety
6. Dick Cacciapouti/Reactor Physics
7. Rudolph Grube/Fuel Management
8. John Rivera/Core Components
9. Francis Quinn/Fuel Procurement, Management & Economic Analysis
10. Jane Grant/Decommissioning Engineering & Services
11. Bruce Holmgren/Decommissioning Engineering
12. Don LeFrancois/Mechanical & Electrical Decommissioning Engineering
13. Bill Henries/Mechanical Engineering Decommissioning Engineering
14. John Bonner/Electrical/I&C Decommissioning Engineering
15. Joe McCumber/Decommissioning Support Services
16. Peter Guglielmino/Vice President Northeast Region
17. Andy Roudenko/Automation Technology
18. Gary McCormick/ Automation Technology
19. Rocky Marcello/Environmental Health & Safety
20. Pete Littlefield/Radiological Engineering
21. Tom Thomas/Radiological Engineering
22. John DiStefano/ Radiological Engineering
23. Andrew Hodgdon/ Radiological Engineering
24. Jake Jacobson/Environmental Services
25. George Harper/ Environmental Services
26. Frank Sabadini/Quality Assurance
27. Cam DiNunzio/Quality Assurance
28. Kathy Bocon/ Human Resources, Training

29. Jackie Copeland/Records Management Services
30. John Gallo/Automation Technology
31. Dean Huggins/Records Management Services
32. Robin Riel/Records Management Services

Documents Reviewed:

1. Nuclear and Fuel Services Organizational Chart
2. Decommissioning Organizational Chart
3. Automation Organizational Chart
4. Environmental Health & Safety Organization Chart
5. QA Organizational Chart
6. Procedure WE-103, Engineering Calculations and Analysis
7. Procedure WE-108, Computer Codes
8. Procedure WE-100, Design Change Request
9. SBC-794, Cycle 6A Loss of Flow Accident Analysis, dated 5/17/97
10. Memorandum, SB Cycle 6 Core Reload Safety Evaluation (Rev 2), dated 10/16/97
11. SB Cycle 6 Core Reload Safety Evaluation, dated 10/14/97
12. SBC-819, Cycle 6 Nuclear Design Report, dated 6/13/97
13. SBC-804, Revised Probable Maximum Hurricane (PMH) Flood Levels
14. VYC-1745, IPEEE Roof Snow Levels
15. Memorandum, MYC-1996 Maine Yankee Post Plant Shutdown Offsite and Control Room Doses for a Fuel Handling Accident dated 9/30/97
16. YC-386, Consequence Evaluation of Vermont Yankee Class 1 Piping in Support of ASME Code Case N-560, dated 1/26/98
17. MYC-1995, Maine Yankee Post Plant Shutdown Maximum Fuel Assembly Radionuclide Inventories dated 9/12/95
18. 97-ENG-01920-C2, Sizing of MP2 Roof Scuppers
19. VY-1627, Diesel Generator Rooms Tornado Venting Analyses, dated 3/7/97
20. Memorandum, ESG50/91, SBP 91-0682

21. Procedures YA-ATD-001, 002, and 003, Customer Support for Software Procedure Development

22. SBC-835, Spent Fuel Thermal Hydraulic Analysis, dated 12/7/97

23. SBC-714, Cycle 5 Nuclear Design Report, dated 11/22/95

24. SBC-718, Cycle 5 RPC Underfrequency LOFA Analysis

25. Procedure NU-3, Work Process

26. NU-11, Task Initiation & Organization

27. W.O. #3010, Structural Evaluation of the Yard Crane & Fuel Building

28. CYC-003, Yard Crane & Fuel Building- Amplified Response Strectra, dated 12/16/97

29. DCP Np. 97-42, Spent Fuel Pool Isolation, dated 12/17/97

30. LPCI/RHR System Notebook-NonQ, Memorandums, System Descriptions, Event Trees, Consequence Analysis

31. Memorandum, SAVY95-020, Review of Diesel Generator CCO Maintenance Plan - No QA Designation, dated 8/25/95

32. SAVY 96-010, Methods for Setting Reliability Performance Criteria for Maintenance Rule, dated 4/30/96

33. Memorandum, SAVY95-015, Risk Importance VY Systems, dated 6/29/95

34. VY-386, Rev 1, Consequence Evaluation of VY Classification, Piping in Support of ASME Code Case N-560

35. Procedure RMS-12, Records Retrieval Indexing/Data Entry

36. QA DYA, Version 3.0/PC Computer Code Modification and Procedure WE-108 Documentation Package - YC-303, Rev 1, dated 4/5/95

37. Computer Code QADUE Migration to IBM PC & Procedure WE-108 Validation, YC-180, dated 6/10/91

38. YRC-1026, Radiological Consequences of a Loss of Shielding Water in the STC

39. Vessel Plasma Cutting Operations Dose Rate to the EAB, Yankee Rowe

40. Memorandum, YAEC Fuel Quality Verification Plan for Vermont Yankee Reload 19, dated 11/7/97

41. Fuel Fabrication Technical Assessment Program for Vermont Yankee Cycle 20/Reload 19, dated 10/97

42. Specification for Nuclear Fuel Assemblies for Maine Yankee Atomic Power Co., dated 2/92

43. Fuel Management Department Procedure and Guideline Manual

44. Mechanical Design Review (MDR) (formerly Engineering Design Review (EDR)) Log, 1996 to Present

45. Mechanical Design Notice (MDN) (formerly Engineering Design Notice (EDN)) Log, 1990 to Present

46. YRC-1062, Reactor Pressure Vessel Shipping Cask

47. YRC-1140, Verification of Stardyne 4.4 for RPV Transportation Calculations

48. WE-008, Standard Memorandum

49. Engineering Instruction No. WE-003, "Indoctrination of Personnel", Revision 13 dated 6/27/97

50. Engineering Instruction No. WE-004, "Training", Revision 12 dated 6/27/97

51. Fuel Management Department Personnel Training Records

52. Appendix R Root Cause Analysis Task Force for Vermont Yankee Report dated September 22, 1995

53. Audit Report No. NSD-96-02 dated December 16, 1996,

    Functional Area Audited: Vermont Yankee Design Engineering

54. Audit Report No. VY-96-25 dated June 17 1996,

    Functional Area Audited: Environmental Qualification (EQ)

55. NRC Integrated Inspection Report 50-271/96-06 dated August 8, 1996,

    Facility: Vermont Yankee Nuclear Power Station

56. Technical Services Agreement Between Maine Yankee and DE&S, 1/2/97

57. Decommissioning Engineering and Services Department Procedure Number 11, Task Initiation and Organization

58. YAEC Records Retrieval Indexing/Data Entry, Records Management Services Procedure RMS-12

59. Decommissioning Work Packages, Procedure No. AP-0103

60. Decommissioning Services Agreement Between YAEC and DE&S, 12/1/97

61. YAEC Technical Administrative Guideline, No., Revision 5, Group Charters

Assessment Report
of
Engineering and Technical Work Processes
Utilized at
DE&S' Bolton Office

Completed

Criteria and Review Approach Document

(CRAD) Worksheets

Criteria and Review Approach Document Worksheet

Assessment Topic No. 1:	**Engineering Process and Adherence to Applicable Procedures, and Industry and NRC Requirements**
Business Process Name:	**Design Systems, Structures, and Components**
Sub-Process Focus:	**Develop Conceptual Design** **Develop Preliminary Design** **Develop Final Design** **Control SSC's Design/Design Basis Changes**

Overall Acceptance Criteria:

Engineering analysis and design processes/activities/products and changes thereto are in accordance with engineering procedures, accepted industry practices, and NRC requirements.

A. Specific Acceptance Criteria:

1. Project task responsibilities are adequately identified.
2. The design products are developed and documented in accordance with procedures.
3. The acceptance criteria are incorporated in the design documents sufficient to allow verification that design requirements have been satisfactorily accomplished.
4. Design requirements are clearly stated.
5. The design product has been adequately verified and
6. Inputs are correctly selected and incorporated into the design.
7. Assumptions are identified and indicated as requiring verification (if needed) when the detailed design activities are completed.
8. Applicable codes, standards and regulatory requirements, including issue and addenda, are properly identified; and requirements for design are met.
9. Past experience with similar designs has been considered.
10. Applicable construction and operating experience are considered.

11. Impact on other design disciplines has been considered.

12. SSC Design interface requirements have been satisfied.

13. Provisions are in place for developing new and revising existing computer models in compliance with regulations

14. An appropriate design method is used.

15. Computer programs are properly used (capabilities, modeling, techniques, program verification).

16. Output is reasonable compared to the inputs.

17. Specified parts, equipment, and processes are suitable for the required application.

18. Specified materials are compatible with each other and the design environmental conditions to which the material will be exposed.

19. Accessibility and other design provisions and features are specified and adequate for performance of needed maintenance and repair, including Inservice Inspection (ISI).

20. The design has properly considered radiation exposure to the public and plant personnel in accordance with ALARA goals and objectives.

21. Adequate pre-operational and subsequent periodic test requirements are appropriately specified.

22. Specifications and other procurement documents adequately specify handling, storage, cleaning and shipping requirements.

23. SSC Identification requirements are adequately specified.

24. Problems or deficiencies that affect work performed in other divisions or sections are considered.

25. A process for initiating and developing design and design basis changes is established, including provisions for reviewing and updating design products.

26. The design product meets design requirements.

27. Revised engineering products have been reviewed and approved by the same organization which reviewed and approved the original documentation.

28. The reason(s) for the revision are clearly and completely stated.

29. All document revisions comply with the same criteria as the original.

30. The design package is complete.

31. The design products are correct.

32. Sufficient review criteria exists and is properly used.

33. Controls are adequate at various design stages.

B. Review Approach:

1. Interview managers responsible for engineering and technical work

2. Interview personnel responsible for producing calculations, analyses and other engineering and technical work products

3. Review pertinent engineering work processes and procedures

4. Review randomly selected engineering and technical work products

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C. Results:

The engineering process and adherence to applicable procedures and industry and NRC requirements were found acceptable. Discussion with a number of managers and engineers, review of procedures, and review of a number of engineering work products were performed to substantiate this result. These reviews confirmed that, for example, design requirements, inputs, assumptions, and code and standard requirements were incorporated in analyses; that appropriate design methods were used; that computer software is properly verified and controlled; and that revisions are properly reviewed and approved. Where appropriate, supplementary work place procedures were developed to assist personnel in implementation of mandatory procedures. Responsibilities were generally well understood.

D. Conclusions:

Procedures were found to be effective and correctly implemented. Engineering and technical work products were also determined to be sound, defensible and in accordance with industry and regulatory requirements. There was clear evidence of a strong culture that is conscious of safety and quality requirements. Pride and ownership of work were also demonstrated. Several ongoing initiatives are targeted on improvement in engineering work processes.

A plan for transitioning from the Yankee organization, work processes and procedures to that of DE&S was not found ready and available for implementation. This has created some deferrals of work, uncertainties in how work will be contracted, and how QA program requirements will be integrated.

The existence of different methods for initiating work and defining QA requirements for work tasks appears to present unnecessary challenges to meeting client expectations. Likewise, different levels of involvement in development of task schedules and in methods for tracking work and progress result in additional unwarranted challenges. There is some indication that unreasonable requests for resources have also had an impact on work process performance.

Specific objective evidence in support of the conclusions include:

1. DE&S Bolton organization chart
2. Technical Services Agreement between Maine Yankee and DE&S dated 1/2/97
3. Decommissioning Work Package Procedure No. AP-0103
4. Decommissioning Services Agreement between YAEC and DE&S dated 12/1/97
5. Nuclear Engineering Department Procedures 1-8
6. Yankee Atomic Technical Administrative Guideline No. 10, Group Charters

Criteria and Review Approach Document Worksheet

Assessment Topic No. 2: **Design Controls, Including Engineering Documentation and Quality of Work Products**

Business Process Name: **Design Systems, Structures, and Components**

Sub-Process Focus: **Control Design Products**

Overall Acceptance Criteria:

Design document controls are applied to work products in accordance with design procedures and QA requirements, including engineering documentation.

A. Specific Acceptance Criteria:

1. Process describes how design documents are controlled in the work place, including documentation of design changes.
2. Requirements for record preparation, review, approval, distribution, retention, etc., are adequately specified.
3. The design product has been entered into the document control system.
4. The design package is complete.
5. The design products are accessible and being used.

B. Review Approach:

1. Review pertinent engineering work processes and procedures.
2. Review randomly selected engineering and technical work products.

C. Results:

Overall, internal work procedures are thorough, mature and contain appropriate industry and regulatory requirements providing good design control methods. Engineering work products have been prepared in accordance with these procedures and are technically correct and judged to be of high quality. Work

packages reviewed were complete including acceptance criteria. A standard memorandum is typically used to transmit engineering work products which summarizes the scope, inputs, results and potential impacts of the deliverable. Some confusion was observed regarding control of engineering work products. This appears to be partially due to the multiple changes in organization, responsibilities and method of interface with the client. A documentation control process was evident which ensures that when work products such as drawings, calculations and analyses are revised, owners of affected documents are notified to enable them to determine any impact and need for appropriate revisions.

D. Conclusions:

Documents are accurate, complete and in accordance with procedures. Procedures adequately describe requirements for review, verification, approval, distribution and records including software development, verification and application. The process for controlling new and revised engineering documents is adequate. The process used for transmitting engineering work products is an effective tool for formally transmitting the deliverable to the client.

Specific objective evidence in support of the conclusions include:

1. Yankee Atomic Electric Company Operational (YAEC) QA Manual, YOQAP-1-A, Revision 27
2. Decommissioning Engineering and Services Department Procedure No. 11
3. YAEC Records Retrieval Indexing/Data Entry, Records Management Services Procedure RMS-12
4. Review of Maine Yankee design package for spent fuel pool island
5. Review of numerous analyses such as YRC-1026, YRC-1062 and validation of QADUE which provide clear requirements, proper analysis and compliance with applicable procedures
6. Review of standard memorandums providing deliverables to clients in accordance with procedures such as Memorandum MYC-1996, Maine Yankee Post Plant Shutdown Offsite and Control Room Doses for a Fuel Handling Accident, and Memorandum Seabrook Cycle 6 Reload Safety Evaluation, Revision 2

Criteria and Review Approach Document Worksheet

Assessment Topic No. 3: **Management Control of Safety-Related Activities**

Business Process Name: **Manage People**

Sub-Process Focus: **Organize Human Resources**
Deploy People

Overall Acceptance Criteria:

Engineering and technical organizations are structured and organized to provide the required skills and abilities to perform safety-related work; and personnel are assigned work based on job knowledge and required training and qualifications.

A. Specific Acceptance Criteria:

1. Organizational responsibilities and lines of authority are defined and understood.
2. Personnel possess the required qualifications, knowledge and skills for the work assigned.
3. The number of personnel in the organization is appropriate for the work load.
4. The organizational structure facilitates conduct of the work.
5. Clear communication lines with clients and regulators are established and adequately maintained as appropriate for the work task.

B. Review Approach:

1. Interview managers responsible for engineering and technical work.
2. Interview personnel responsible for producing calculations, analyses and other engineering and technical work products

C. Results:

The organization is designed to provide engineering and technical support in the areas of nuclear engineering and fuel services, decommissioning, environmental health & safety and QA to the sponsor

plants; that being Maine Yankee, Vermont Yankee, Yankee Rowe, and Northeast Utilities (Millstone, Connecticut Yankee, and Seabrook). Managers and engineers were very knowledgeable of their technical responsibilities but some appeared to be less clear of their functional roles and responsibilities and interface with the sponsor plants. However; roles and responsibilities appear to be evolving with company and client organizational and ownership changes and are different depending on the client. Personnel typically have 20+ years of experience in their engineering disciplines which provides depth for required independent review and verification of work products. The number of personnel in the organization appeared to be marginally adequate for the work load. Concerns over workload and schedule pressures were prevalent through much of the organization. The use of outside resources has been necessary on occasions in order to meet deliverable schedules.

A close working relationship was found to exist between engineering groups and their clients. These relationships in many cases have existed for many years and are very strong. Depending on the plant, some groups are more formal than others in dealing with the client on work requests and applicable QA requirements. In some groups work is initiated with little or no documentation. However; these relationships are now evolving due to the company and plant organizational changes. As one example, the Projects group use to provide support between the clients and the engineers relative to project assignments. However, now they are located at the site and no longer provide direct support. This has led to some confusion as to how work is formally initiated.

D. Conclusions:

The organizational structure is appropriate for the engineering functions performed for the sponsor plants. Staffing levels appear to be marginally adequate. Outside support is occasionally used to meet schedules. Personnel are professional and very knowledgeable of their job. Roles and responsibilities from a business point of view (i.e. budget and schedule type issues) are evolving and are different for each project/plant. Consequently, not everyone appears to have a good understanding of these. Interfaces and lines of communication between the engineering groups and the plants are likewise changing and not clearly documented.

Specific objective evidence supporting the above conclusion was found in the following documents:

1. DE&S Bolton organization chart
2. Yankee Atomic Electric Company Operational (YAEC) QA Manual, YOQAP-1-A, Revision 27
3. Yankee Rowe Station Decommissioning Engineering Status dated 1/9/98
4. EH&S Task/Status List dated 2/2/98
5. Nuclear Engineering Project List Status List/Biweekly report dated 2/2/98
6. ATD Work Status List dated 1/21/98
7. Client Status Reports:
   - 1/26/98 Status/Letter to WR Carr/NU
   - 11/21/96 Status/Letter to JE Kmietek
   - 11/13/95 Status/Letter to EA DeBarba

Criteria and Review Approach Document Worksheet

Assessment Topic No. 4: **Design Engineering Training; Particularly, QA Program Training**

Business Process Name: **Manage People**

Sub-Process Focus: **Train People**

Overall Acceptance Criteria:

Personnel assigned work are trained and qualified based on training needs assessments to perform required tasks in accordance with design procedures and QA requirements.

A. Specific Acceptance Criteria:

1. An employee training needs assessment process is established and utilized.
2. There is a capability and process to develop and provide employee training.
3. There is a process for keeping up with the training achieved by employees and the associated training records.
4. There is a process for identification of periodic training needs of employees.
5. There is a process to evaluate training effectiveness which is provided to the organization.
6. On-the-job training (OJT) is adequately integrated into the overall approach to training.

B. Review Approach:

1. Assess the adequacy and effectiveness of implementation of the Indoctrination and Training Program.
2. Interview responsible department and training personnel.
3. Review Indoctrination and Training Records for randomly selected individuals.

C. Results:

Indoctrination of engineering personnel is governed by Engineering Instruction (EI) No. WE-003, "Indoctrination of Personnel". The purpose of this document is to:

1. provide a program for the indoctrination of engineers who are responsible for accomplishing engineering activities in accordance with the Engineering Manual. This procedure is also used for the indoctrination of contractors working under YAEC's Quality Assurance Program.

2. establish how the content and completion of the indoctrination program for engineers is documented.

Documentation of the program is accomplished by the use of the Indoctrination of Personnel Form (WE-003-1). Based on a review of the Indoctrination Records for the individuals identified below, this procedure is being effectively implemented. In particular, it was verified that EI Nos. WE-103, "Engineering Calculations and Analysis" and WE-108, "Computer Codes" were reviewed in detail and that the Project, Engineering or Group Manager reviewed the completed checklist.

Employee Name	Organization
R. Cacciapouti	Nuclear Engineering
A. Fyfe	Nuclear Engineering
J. Distefano	Environmental Health & Safety
J. Gorski	Nuclear Engineering
J. Hamawi	Environmental Health & Safety
G. Harper	Environmental Health & Safety
P. Littlefield	Environmental Health & Safety
K. Spinney	Nuclear Engineering
M. Tremblay	Nuclear Engineering

However, the Indoctrination of Personnel Form for G. McCormick from Automation Technology could not be located in the Document Control Center. When this was brought to the attention of the Responsible Manager, immediate corrective measures were taken. Specifically, the Indoctrination of Personnel Form was completed, provided evidence of an appropriate level of understanding of WE-103 and WE-108, and turned over to Records Management Services for proper filing and storage. It was concluded that this was an isolated case based on the absence of similar findings.

In addition to Indoctrination Records, the Training Records for these individuals were also reviewed and found to be in accordance with the requirements of EI No. WE-004, "Training". These records demonstrated that the training program ensures that employees and contractors receive periodic updating in the requirements of the program and the experience accrued in the implementation of the Engineering Manual. In particular, the Education Profile Forms, which are maintained by the Training Department, were reviewed. This form documented the internal and external training courses, sessions, etc. that were attended. Updates to this document were based on input from the Functional Manager.

D. Conclusions:

Based on a review of EI Nos. WE-003, "Indoctrination of Personnel" and WE-004, "Training", interviews with responsible department and training personnel, and a review of the Indoctrination and Training Records for randomly selected individuals, the Indoctrination and Training Program was found to be adequate.

However, Condition Report (CR) No. 97-0077 dated July 25, 1997, Action Item No. 08 required that an assessment be performed by each department to evaluate if the read and sign training performed was appropriate to assure adequate understanding of material. The response to Action Item No. 08 was that the read and sign training was not an appropriate method of training and would no longer be used. Further, to ensure that employees unable to attend training received adequate training, a review packet containing a quiz would be developed as part of a Lesson Plan. Based on discussions with the Training Department, it was learned that the Functional Managers identified selected EI and Technical Administrative Guidelines (TAG) for this type of training. Presently, CR No. 97-0077 remains open. Although enhanced training was completed for TAG No. 14, "Policy for Control of Safeguards Information" on December 17, 1997, no additional training has occurred to date. This initiative has been placed on hold pending completion of the YAEC/DE&S transition.

Criteria and Review Approach Document Worksheet

Assessment Topics No. 5: **Management Control of Safety-Related Activities; Adherence to Applicable Procedures, and Industry and NRC Requirements; and Quality of Work Products**

Business Process Name: **Manage Performance**

Sub-Process Focus: **Monitor/Review/Assess Performance**

Act on Performance Results

Overall Acceptance Criteria:

Performance is monitored, assessed and appropriate actions taken based on results.

A. Specific Acceptance Criteria:

1. Acceptable standards of performance are established.
2. The organization understands the acceptable standards of performance.
3. There is a process for collecting and evaluating performance.
4. Corrective actions are prompt and adequate.
5. Management control and oversight of safety-related activities are adequate.

B. Review Approach:

1. Interview managers responsible for engineering and technical work.
2. Review randomly selected internal and external assessments (audits, analysis and inspections). Assess the results of these oversight activities with respect to the significance and impact of the issues identified, corrective actions taken and their closure.

C. Results:

Management control of safety-related activities, adherence to applicable procedures, and quality of work products were found acceptable. Interviews with a number of management personnel indicated good knowledge of the work in house and the challenges facing the organization. The review of work products and discussion at all levels in the organization showed that work is adequately controlled and is prepared in accordance with accepted standards. Consequently, based on the evidence of satisfactory work before it and the motivated, experienced work force which we found, the team did not find reason to pursue some of the traditional aspects of performance management in detail. For example, acceptance criteria 1-3 were not evaluated for these reasons.

In a number of interviews, the review team discussed with management its perspective regarding protecting the acceptable standards of work performance from the pressures of schedule. In all cases, management held out work product quality as the number one priority. The review team did not discover any instances in which work product quality had suffered based on the review of the specific deliverables. It was also evident that the Bolton office is developing methods to address the concerns over schedule pressure. Managers in all departments assessed were found to be aware of work assignments and directly involved in many cases as the approver for safety-related work.

In addition, a review was conducted of randomly selected internal and external assessments such as audits, analysis and inspections. Four oversight activities were chosen from twenty-five potential audits/assessments performed by Quality Assurance of YAEC/YNSD Engineering activities. The results of these oversight activities were assessed with respect to the significance and impact of the issues identified, corrective actions taken and their closure. The results of this review are documented below:

- Appendix R Root Cause Analysis Task Force for Vermont Yankee

  Audit Dates: August 28 - September 15, 1995

  The Task Force performed a root cause analysis on six Vermont Yankee (VY) event reports involving the VY fire protection program and compliance to Appendix R requirements. Several recommended corrective actions were implemented and assigned for plant commitment tracking. An aggressive Task Force was established by the VY Plant which addressed the identified wide range of issues. Five of the seven items have been satisfactorily closed. The two remaining issues are being tracked to closure through Licensee Event Report (LER) No. LER9514 05. The plant and YNSD Project Engineering have re-organized the Fire Protection/ Appendix R coordination, and greatly improved and continues to improve the Appendix R Program in meeting all of the established federal requirements.

- Audit No. NSD-96-02, Functional Area Audited: VY Design Engineering

  Audit Dates: November 4-8, 11-13, and 22, 1996

  The Audit Team verified the implementation of the Quality Assurance Program for design and design change controls and confirmed the adequacy of the program in meeting regulatory

requirements and internal YNSD engineering procedures. It was concluded that design activities performed by the VY Design Engineering Group were considered adequate. Engineering Design Change Requests (EDCRs) and calculations reviewed by the audit technical specialists in the electrical/mechanical disciplines identified no immediate design or operability issues. However, programmatic issues were identified by the audit team, which required VY Engineering Management attention. Thirteen findings were identified and five recommendations for improvement were offered. Nine of the thirteen findings have been satisfactorily closed. The four remaining findings are being tracked to closure through ER Nos. 96-1074, 96-1083, 96-1088 and 96-1090. In addition, all five recommendations were satisfactorily closed.

- Audit No. VY-96-25, Functional Area Audited: Environmental Qualification (EQ)

  Audit Dates: April 22 - May 10, 1996 and May 16, 1996

  The EQ Design Basis Audit Team verified that a solid design basis was established and maintained for the VY EQ Program, and that it met the requirements of 10CFR50.49 and guidance provided by Regulatory Guide 1.89. The audit consisted of the review of EOPs, calculations regarding High Energy Line Breaks (HELBs) outside containment, specifically large circumferential breaks (double-ended), and small break analysis for the Reactor Water Clean Up (RWCU) System and steam tunnel, Loss-Of-Coolant Accidents (LOCAs) inside containment, procurement of EQ equipment, EQ manuals, NRC generic letters and bulletins, prior submittals to the NRC (SERs), NRC Memorandum and Orders (M&Os), radiation dose calculations and analysis associated with the DBAs, extensive interviews, configuration control, and review of FSAR for EQ applicability. The audit concluded that the EQ Program at VY was supported by an adequate basis. However, the audit team determined that although the documentation reviewed was technically adequate, increased management attention was warranted in the level of detail needed in certain EQ Program basis documentation and configuration control within the EQ Program (control of EQ changes affecting operations and EOPs). As a result of the audit, five findings were identified and five recommendations for improvement offered for consideration. Two of the five findings have been satisfactorily closed. The three remaining findings are being tracked to closure through ER Nos. 96-0277, 96-0293 and 96-0292. In addition, four of the five recommendations have been satisfactorily closed.

- NRC Integrated Inspection Report No. 50-271/96-06 dated August 8, 1996

  Inspection Dates: May 12 - July 13, 1996

  Licensee: Vermont Yankee Nuclear Power Corporation

  Facility: Vermont Yankee Nuclear Power Station

  The NRC performed an Integrated Inspection at the Vermont Yankee Nuclear Power Station. This Integrated Inspection included aspects of Licensee Operations, Engineering, Maintenance and Plant Support. The report covered a nine week period of Resident Inspection and the results of announced inspections by a Regional Specialist and a Regional Projects Inspector.

The NRC Integrated Inspection Team concluded that the conduct of activities at the Vermont Yankee Nuclear Power Station were generally characterized by safety-conscience operations, sound engineering and maintenance practices and careful radiological work controls. It was noted that the Vermont Yankee staff continues to demonstrate a questioning attitude in identifying program discrepancies and/or old design problems such as the Appendix J Valve Testing Oversight and the Emergency Diesel Generator Room Differential Protection Inadequacies discovered during the NRC Integrated Inspection. In the area of Plant Support, the NRC Inspection Team found that Vermont Yankee continued to implement overall effective Radiological Environmental Monitoring and Meteorological Monitoring Programs. Lastly, no Notice of Violations (NOVs) or Nonconformances were issued by the NRC as a result of the Integrated Inspection nor was a reply to the NRC Integrated Inspection Report No. 50-271/96-06 required.

D. Conclusions:

Based on interviews with management, reviews of engineering work products and reviews of audits, it was concluded that management control and oversight of safety-related work is acceptable. No significant quality or technical issues were identified.

Specific objective evidence in support of these results include the Yankee Atomic Electric Company Operational (YAEC) QA Manual, YOQAP-1-A, Revision 27. These QAP's provide management with necessary controls in accordance with NRC criteria for assuring design control.

APPENDIX G

QUALITY ASSURANCE ASSESSMENT

DUKE ENGINEERING & SERVICES
ASSESSMENT PROCESS REVIEW REPORT

Process Review Dates: January 26 -- February 6, 1998

Organization Reviewed: Quality Assurance Services Department in Bolton, MA [formerly part of the Yankee Atomic Electric Company (YAEC) Nuclear Services Division]

Business Unit Manager: C.R.Clark

Activities Reviewed: The assessment process (audits, surveillances, self-assessments) implemented by Yankee Atomic Electric Company for 1995, 1996, and 1997.

Team Leader: D.A.WALKER / D.A.Walker / 2/16/98
Printed Name/Signature/Date

Approval: M L SANGER / ML Sanger / 2/16/98
QA Manager/Signature/Date

Report Date: 2/16/98

1.0 PURPOSE

As stated in the Assessment Process Review Team (APRT) Charter, Revision 0 (Attachment 1 to this report), the objective was to validate that assessment activities (audits, surveillances, management assessments, etc.) were in accordance with accepted industry practices, specifically:

- Assessments demonstrated a regard for and adherence to applicable industry/NRC requirements;
- Assessments were performed in accordance with approved procedures by qualified individuals;
- Individuals leading assessments were appropriately trained and knowledgeable of the QA program,
- Assessment documentation was accurate and complete;
- Management involvement in the assessment process was evident; and
- Assessments covered both programmatic requirements and technical work processes.

2.0 SCOPE

The APRT's scope included a review of assessment related documentation completed over the past three years by the YAEC Quality Assurance Services Department (formerly part of the YAEC Nuclear Services Division).

The APRT had complete access to all company personnel and documentation. For the purpose of this review, company personnel included, but were not limited to, employees, suppliers, subcontractors, consultants, and temporary employees. The APRT utilized a combination of document reviews and personnel interviews. Documents that were of primary interest to the APRT included assessment procedures, qualification records for assessment team members, QA training records for assessment team leaders, assessment reports, assessment checklists (when applicable), and assessment corrective action and close-out documentation. See Section 5 of this report for specific areas reviewed.

3.0 TEAM

The APRT team consisted of the Team Leader and two Quality Assurance individuals. The team members were independent of the assessments reviewed. The APRT team and assignments follow:

- David Walker, DE&S Team Leader:
  - Responsible for ensuring that team members reviewed the Assessment Process
  - Review Charter and understood their assignments.

- Responsible for requesting documentation to be reviewed and coordination of review effort.

- Reviewed QA Training of assessment team leaders to the YAEC QA Program.

- Reviewed qualifications of assessment team members to ensure compliance with assessment procedures.

- Reviewed assessment reports to ensure the following:

  > Management involvement in the assessment process
  > Depth of the assessments (programmatic and technical work processes covered).

- Responsible for preparation of this report.

• Jonathan Gerson, DE&S Team Member, reviewed:

  - Assessment reports to ensure procedural compliance including:

    > A selection of assessment checklists, when applicable, and
    > A selection of assessment corrective action documentation to ensure appropriate close-out.

• Phil Horsmon, DE&S Team Member, reviewed:

  - The QA Manual sections on assessments and assessment procedures for compliance with regulatory requirements.

Team Experience (See Attachment 5 for details.)

Mr. Walker has over twenty-five years of Quality Assurance experience in the manufacturing, construction, and operation phases of commercial nuclear power projects. Sixteen years were with management/consulting responsibilities and eight years as a Quality Assurance Manager.

Mr. Gerson has over twenty-three years of experience in the nuclear field. He has held positions as Quality Assurance Engineer, Technical Analyst, Quality Control Engineer and Procedure Writer. He is experienced in Quality Assurance, procedure writing, training coordination, Quality Control, audit status tracking, procurement documentation reviews, and receipt inspection. Mr. Gerson has also assisted in the development of planning and integration strategies and performance milestone assessments.

Mr. Horsmon has over twenty-one years of experience in the nuclear field. He has held positions as Technical Specialist, QC Inspection Supervisor, QC Planning Engineer, and Senior QC Inspection. He is experienced in the field of nuclear power plant construction and inspection activities as well as Quality Assurance programmatic requirements related to 10CFR50 Appendix B and NQA-1.

4.0 PERSONNEL CONTACTED

This review was conducted from DE&S's Charlotte, N.C. office; therefore, the Team Leader obtained documentation from personnel located in the DE&S Bolton, MA office. Most of the documentation was provided by Janet Allen. Other personnel contacted during the review were:

Steve Bibo
David Karr
Walter Peterson
Frank Sabadini
Steve White

5.0 REVIEW SUMMARY

Review QA Manual Section on Assessments and Assessment Procedures for Compliance with Regulatory Requirements

The following is a summary of documents reviewed for compliance to regulatory requirements:

Operational QA Manual YOQAP-1-A, Rev. 27

The following sections of this manual were reviewed:

- Policy Statement
- Section I, "Organization"
- Section II, "Quality Assurance Program"
- Section XVIII, "Audits"

As a result of this review, there is one observation in regard to the Policy Statement. The Policy Statement addresses 10CFR50 Appendix B as the governing document. Section I of the manual references ANSI N18.7-1976 in addition to 10CFR50 Appendix B (see Observation No. 1).

Yankee Atomic Quality Assurance Procedures Manual

- OQA-XVIII-1, Rev. 9, "Auditor Training and Qualification"

  This procedure was reviewed for regulatory compliance against 10CFR50 Appendix B, Criterion I, II, and XVIII and ANSI N45.2.23-1978, and was found satisfactory.

- OQA-XVIII-2, Rev. 27, "Audit Program"

  This procedure was reviewed for regulatory compliance against ANSI N45.2.12-1977. One observation was noted during this review. Section 5.0 of ANSI N45.2.12-1977 requires audit and personnel records to be collected, stored, and maintained in accordance with ANSI N45.2.9. There is no reference to this requirement in the procedure (see Observation No. 2).

Yankee Atomic Quality Assurance Procedures Manual

- OQA-XVIII-15, Rev. 1, "Internal Audit and Surveillance Program"

  This procedure was reviewed for regulatory compliance against ANSI N45.2.12-1977, although there is no reference made to this standard in the procedure (see Observation No. 3). Assuming that the procedure was prepared in accordance with ANSI N45.2.12-1997, there were no issues.

  This review area is considered satisfactory.

Review QA Training of Assessment Team Leaders to YAEC QA Program

QA training records were reviewed for twenty-three Lead Auditors and five Auditors (see Attachment 2 for listing of names). Qualified Lead Auditors are required to lead audits and surveillances per YAEC procedures and both Lead Auditors and Auditors are required to have QA training. This review resulted in one finding. The finding identifies that orientation training records to applicable standards and QAD procedures could not be located for one Lead Auditor and four Auditors (see Finding No. 1 for details).

Self-assessments per Yankee Atomic Electric Company Technical Administrative Guideline No. 22, Revision 2, "Self Assessment Program," does not address QA training requirements for team leaders/members; therefore, none were reviewed.

With the exception of Finding No. 1, this review area is considered satisfactory.

Review Qualifications of Assessment Team Members to Ensure Compliance with Assessment Procedures

Qualification records for twenty-three Lead Auditors, five Auditors, and thirty-three Technical Specialists were reviewed (see Attachment 2 for listing of names). Per YAEC procedures, team members participating in audits and surveillances are required to have a qualification record. This review resulted in one finding. The finding identifies incorrect qualification records for five Lead Auditors, missing qualification records for five Auditors, and missing qualification records for seven Technical Specialists (see Finding No. 1 for details).

Yankee Atomic Electric Company Technical Administrative Guideline No. 22, Revision 2, "Self-Assessment Program," does not address special qualification requirements for assessment team members; therefore, none were reviewed.

With the exception of Finding No. 1, this review area is considered satisfactory.

Review Assessment Reports to Ensure Procedural Compliance Including a Selection of Assessment Checklists and a Selection of Assessment Corrective Action Documentation to Ensure Appropriate Close-out

The following were reviewed covering assessments from 1995 through 1997 (see Attachments 3 and 4 for details):

- 25 Audit Reports including
  - 9 for audit plans and checklists
- 25 Surveillance Reports
- 115 Summary and Status of Corrective Action (SSCA) Reports including
  - 76 from inception to closure
- 24 Event Reports (ER)
- 37 Condition Reports (CR)
- 1 Vendor Surveillance Finding Report (VSFR)
- 1 Root Cause Analysis

The review noted that the YAEC assessments (audits, surveillances, self-assessments, etc.) followed their respective procedure requirements, and were thorough in their investigations. The reviewed documentation was accurate, complete and was detailed enough to be able to follow the assessment process from start to finish.

Four observations were identified during this portion of the review. The details are in Section 7.0, "Observations." Observation No. 4 involves surveillances which identify "unsats" without any form of findings generated. Observation No. 5 involves an undocumented mechanism to extend due dates for CRs. Observation No. 6 identifies what appears to be a high overdue rate of SSCAs. Observation No. 7 involves numerous self-assessment findings.

This review area is considered satisfactory.

Review Assessment Reports to Ensure Management Involvement and the Depth of the Assessments (Programmatic and Technical Work Processes Covered)

Twenty-five audit reports, twenty-five surveillance reports, and two Functional Area Assessment reports (1996 and1997) were reviewed to ensure management involvement and the depth of assessments (see Attachments 3 and 4 for complete listings of reports reviewed).

Management involvement was evident in all three types of assessments based on the distribution lists for reports. Audit and surveillance reports were distributed to the YAEC President, responsible Vice-Presidents, and responsible management. The two Functional Area Assessments were provided to the YAEC President. Management involvement was also evident by reviewing

entrance and exit meeting attendees lis[illegible]s and by reviewing corrective action responsibilities for both audits and surv[illegible]

The depth of assessments is adeq[illegible]rogrammatic criteria applicable to Engineering activities were cover[illegible]di[illegible]

- Organization
- QA Program
- Design Control (including computer software controls)
- Instructions, Procedures, and Drawings
- Document Control
- Corrective Action
- Quality Assurance Records

In addition, Procurement was covered at least annually. Due to potential conflicts, Audits was covered annually by Joint Utility Management (JUMA) audits.

Functional areas were covered on an annual basis by audits and included reviews of technical work processes. Technical Specialists were used on audits to perform technical work process reviews and identify deficiencies in technical areas (see Attachment 3 for examples).

This review area is considered satisfactory.

6.0 FINDINGS

Finding No. 1

This Finding involves missing or incorrect documentation supporting the qualifications and training of certain individuals who participated in audits and surveillances reviewed. Items listed are contrary to Yankee Nuclear Services Division Quality Assurance Department Procedure OQA-XVIII-1, Revision 9, "Auditor Training and Qualification."

Part A - Lead Auditors

Alan M. Abrahamovich - Contrary to Section IV.B.2.c, the five audits listed on the Record of Lead Auditor Qualification were dated in 1995 after the original certification date of 1/15/95. (Note: A corrected copy of Mr. Abrahamovich's qualification record, changing the audit dates to 1994, was provided to the Team Leader during this review. The corrected dates were matched against dates in the reports for the audits in question and were found to be correct. This was a typing error and no further action is required.)

James E. Behm - Contrary to Section IV.B.2.a, the Record of Lead Auditor Qualification indicates 9 total points instead of the required 10.

Stephen L. Bido - Contrary to Section IV.B.2.c, only two of ten audits listed on the Record of Lead Auditor Qualification had been performed in the three year period prior to the date of certification as required.

Michael P. Saniuk - Contrary to Section IV.B.2.c, only four of seven audits listed on the Record of Lead Auditor Qualification had been performed in the three year period prior to the date of certification as required.

Allen E. Winegard - Contrary to Section IV.B.2.c, only four of ten audits listed on the Record of Lead Auditor Qualification had been performed in the three year period prior to the date of certification as required.

Allen E. Winegard - Contrary to Section IV.B.1.a, orientation training records to applicable standards could not be located for this individual.

Note: See Attachment 2 for list of audits/surveillances performed by the Lead Auditors.

Part B - Auditors

J.Buck, J.Frothingham, D.Head, C.Rowe, and J.Marchi - Contrary to Section IV.C.2, auditor qualification records (Form OQA-XVIII-1.3) could not be located for these individuals.

J.Frothingham, D.Head, C.Rowe, and J.Marchi - Contrary to Section IV.B.1.a, orientation training records to applicable standards and QAD procedures could not be located for these individuals.

Note: See Attachment 2 for list of audits/surveillances participated in by the Auditors.

Part C - Technical Specialists

Contrary to Section IV.C.3, Technical Specialist Orientation, Training, and Instruction forms (Form OQA-XVIII-1.4) could not be located for the following individuals/audits:

Technical Specialist/(Audit Number)
J.Dallman (NSD-97-17)
D.Danielski (NSD-97-17)
C.Everline (NSD-97-17)
E.Ferat (NSD-97-17)
G.House (NSD-97-06)
K.Michalski (NSD-97-06)
R.Petrone (NSD-96-04)

7.0 OBSERVATIONS

Observation No. 1

The Policy Statement in Operational QA Manual YOQAP-I-A, Rev. 27 addresses 10CFR50 Appendix B as the governing document. Section I of the manual references ANSI N18.7-1976 in addition to 10CFR50 Appendix B.

Observation No. 2

Section 5.0 of ANSI N45.2.12-1977 requires audit and personnel records to be collected, stored, and maintained in accordance with ANSI N45.2.9; however, this requirement is not referenced in OQA-XVIII-2, Rev. 27, "Audit Program."

Observation No. 3

It is assumed that ANSI N45.2.12-1997 was used as a basis for preparation of procedure OQA-XVIII-15, Rev.1, "Internal Audit and Surveillance Program." However, the standard is not referenced in the procedure.

Observation No. 4

Some surveillances identified "unsats," but did not generate any deficiency reports or a clear explanation as to why none were generated.

Observation No. 5

It was noted that CRs do not have a documented mechanism to extend due dates (responses, corrective action completions, etc.).

Observation No. 6

Of the 76 SSCAs followed to completion, approximately 21% were overdue at one stage of the process.

Observation No. 7

Seventeen deficiency reports (SSCAs, CRs & ERs) were generated from May 1996 through November 1997 dealing with self-assessments over a wide area of departments and some repeat areas.

8.0 CONCLUSION STATEMENT

Overall, the assessment process implemented by Yankee Atomic Electric Company for 1995, 1996, and 1997 was found to be of high quality and in compliance with regulatory requirements and internal procedure commitments. The review resulted in one finding. Although the finding is not considered significant, it does identify missing or incorrect documentation supporting the qualifications and training of certain individuals who participated in the audits and surveillances reviewed. It is recommended that the documentation identified in the finding be corrected and a corrective action response provided to the Review Team Leader by March 2, 1998. The response should address the extent of the problem identified and any impact the finding may have on the audits and surveillances. Seven observations were also identified which do not require a response.

APPENDIX G

ATTACHMENT 1

ASSESSMENT PROCESS REVIEW TEAM CHARTER

CHARTER

ASSESSMENT PROCESS REVIEW TEAM

Purpose

The Assessment Process Review Team's (APRT) objective is to validate that assessment activities (audits, surveillances, management assessments, etc.) are in accordance with accepted industry practices, specifically:

- assessments demonstrate a regard for and adherence to applicable industry/NRC requirements;
- assessments were performed in accordance with approved procedures by qualified individuals;
- individuals leading assessments were appropriately trained and knowledgeable of the QA program;
- assessment documentation is accurate and complete;
- management involvement in the assessment process is evident; and
- assessments cover both programmatic requirements and technical work processes.

Scope

The APRT's scope includes a review of assessment related documentation completed over the past three years by the Quality Assurance Services Department (formerly part of the YAEC Nuclear Services Division).

The APRT shall have complete access to all company personnel and documentation. For the purpose of this Charter, company personnel include, but are not limited to, employees, suppliers, subcontractors, consultants, and temporary employees. The APRT will utilize a combination of document reviews and personnel interviews. Documents that are anticipated to be of primary interest to the APRT include assessment procedures, qualification records for assessment team members, QA training records for assessment team leaders, assessment reports, assessment checklists (when applicable), and assessment corrective action and close-out documentation.

Reviews shall include the following as a minimum:

1. Review QA Manual section on assessments and assessment procedures for compliance with regulatory requirements.

2. Review QA Training of assessment team leaders to YAEC QA Program.

3. Review qualifications of assessment team members to ensure compliance with assessment procedures.

4. Review assessment reports to ensure procedural compliance and the following:

   - management involvement in the assessment process
   - depth of the assessments (programmatic and technical work processes covered)

5. Review selection of assessment checklists, when applicable, to ensure procedural compliance.

6. Review selection of assessment corrective action documentation to ensure procedural compliance and appropriate close-out.

Team

The APRT team consists of the Team Leader and two Quality Assurance individuals. The team members are independent of the assessments being reviewed. The APRT team is:

- David Walker, Team Leader, DE&S
- Jon Gerson, Team Member, DE&S
- Phil Horsmon, Team Member, DE&S

The Team Leader is responsible for ensuring that team members have reviewed this Charter and understand their assignments. Technical Specialists, independent of the issues being reviewed, may be called upon to determine technical adequacy, if determined necessary by the Team Leader.

Deliverables

The APRT shall prepare a stand alone report that, as a minimum:

1. Describes the purpose and scope of the review.

2. Describes the background of each APRT member and the role assigned to each during the review process.

3. Identifies personnel contacted during the review process.

4. Identifies documents reviewed (including document number, date, title, and document type).

5. Describes findings, recommendations, observations, and includes a conclusion statement which addresses the objectives listed in the Purpose section of this Charter.

Schedule

The APRT schedule is:

1/26/98 - Begin review
1/30/98 - End review
2/6/98 - Issue final report

M.L.Sanger, D&E&S QA Manager

ASSESSMENT PROCESS REVIEW DOCUMENTATION PERSONNEL QUALIFICATIONS					
Name	**Level**	**Lead Auditor Qualified**	**Comments**	**Audits Performed (L=Lead)**	**Surveillances Performed (L=Lead)**
Buck, J.	Auditor			NSD-96-06	
Frothingham, John	Auditor			NSD-96-01	
Head, D.	Auditor				NSD-97-01-01
Rowe, C.	Auditor			NSD-96-16	
Marchi, J.	Auditor			NSD-95-03	
Abrahamovich, Alan M.	Lead Auditor	1/99 - 1/15/95		NSD-95-06, NSD-96-06 (L), NSD-97-17, NSD-97-18	NSD-97-02-01 (L)
Bateman, S. W.	Lead Auditor	1/99 - 3/17/97		NSD-96-06, NSD-96-12, NSD-97-10, NSD-97-17, NSD-97-18 (L)	NSD-97-01-02 (L), NSD-97-02-02 (L), NSD-97-04-01 (L), NSD-96-03-01 (L), NSD-97-13-02
Bearham, Fred	Lead Auditor	5/98- 5/1/97			NSD-97-07-03, NSD-97-14-01 (L)
Behm, James E.	Lead Auditor	1/98 - 11/4/96			NSD-96-03-01 (L)
Bibo, Stephen L.	Lead Auditor	1/99 - 7/1/96		NSD-97-13 (L)	NSD-97-01-01 (L)
Bourassa, Joseph F.	Lead Auditor	1/99 - 12/20/89		NSD-95-12 (L), NSD-96-08	
Bozek, Lucas X.	Lead Auditor	1/99 - 1/8/79	Evaluated every two years between 1/8/79 and 1/3/83	NSD-96-04 (L)	
Childress, T. A.	Lead Auditor	1/99 - 3/11/85		NSD-95-13 (L)	
DiNunzio, Cabillo A.	Lead Auditor	1/99 - 4/21/95		NSD-95-01, NSD-95-07, NSD-96-01 (L), NSD-96-04, NSD-96-07 (L), NSD-96-08 (L), NSD-96-15 (L), NSD-96-16 (L), NSD-97-06, NSD-97-10 (L), NSD-97-13, NSD-97-17 (L), NSD-97-18	NSD-97-07-01 (L), NSD-97-07-03 (L), NSD-97-11-01 (L), NSD-97-13-01 (L), NSD-97-13-02 (L)

ASSESSMENT PROCESS REVIEW DOCUMENTATION PERSONNEL QUALIFICATIONS					
Name	**Level**	**Lead Auditor Qualified**	**Comments**	**Audits Performed (L=Lead)**	**Surveillances Performed (L=Lead)**
Hall, Bruce. E.	Lead Auditor	1/99 - 10/31/97		NSD-97-17, NSD-97-18	NSD-97-02-04 (L), NSD-97-07-02 (L), NSD-97-07-03, NSD-97-12-01 (L), NSD-97-13-02
Homer Jr., Everett	Lead Auditor	1/99 - 6/13/97		NSD-97-17	
Karr, David P.	Lead Auditor	1/99 - 8/23/96		NSD-96-12 (L), NSD-96-13, NSD-97-17	
Martin, R. L.	Lead Auditor	1/99 - 1/20/76	Evaluated every two years between 1/20/76 and 1/11/78, not evaluated again till 1/2/81	NSD-95-06 (L), NSD-95-07B (L)	
Peterson, Walter K.	Lead Auditor	1/99 - 8/1/94		NSD-95-07	
Ramsdell, Roy A.	Lead Auditor	1/99 - 7/8/94			
Sabadini, Frank H.	Lead Auditor	1/99 - 10/2/86		NSD-95-02 (L), NSD-95-04 (L), NSD-95-07, NSD-96-01, NSD-96-02 (L), NSD-96-15, VY-96-25 (L), NSD-97-06 (L), NSD-97-17	NSD-97-02-03 (L), NSD-97-04-02 (L), NSD-97-06-01 (L), NSD-97-06-02 (L), NSD-97-06-03 (L), NSD-97-09-01 (L), NSD-97-15-01 (L)
Saniuk, Michael P.	Lead Auditor	1/99 - 7/6/96		VY-96-25	
Sychevsky, Anthony	Lead Auditor	1/99 - 9/9/93			
Syrell, Thomas J.	Lead Auditor	1/98 - 12/4/95		NSD-95-03 (L), NSD-96-13 (L)	
Walulik, Robert J.	Lead Auditor	1/99 - 1/13/97		NSD-96-01, NSD-96-12	NSD-97-08-01 (L)
White, Steven C.	Lead Auditor	1/99 - 1/9/84		NSD-95-07 (L)	
White, Thomas P.	Lead Auditor	1/98 - 3/2/95		NSD-95-01 (L)	
Winegard, A. E.	Lead Auditor	1/97 - 11/7/95		NSD-96-10 (L)	

ASSESSMENT PROCESS REVIEW DOCUMENTATION PERSONNEL QUALIFICATIONS					
Name	**Level**	**Lead Auditor Qualified**	**Comments**	**Audits Performed (L=Lead)**	**Surveillances Performed (L=Lead)**
Barth, Andrew	Tech Specialist			NSD-96-01	
Blaisdell, J.	Tech Specialist			VY-96-25	
Dallman, J.	Tech Specialist			NSD-96-16, NSD-97-17	
Danielski, D.	Tech Specialist			NSD-97-17	
Desilets, J.	Tech Specialist			VY-96-25	
Distefano, J.	Tech Specialist			VY-96-25	
Everline, C.	Tech Specialist			NSD-97-17	
Ferat, E.	Tech Specialist			NSD-97-17	
Gerlits, David	Tech Specialist			NSD-96-01	
Harris, E.	Tech Specialist			NSD-96-06	
Hengerle, G.	Tech Specialist			NSD-95-01	
House, G.	Tech Specialist			NSD-97-06	
Jones, P	Tech Specialist			NSD-96-12	
Labrecque, N.	Tech Specialist			NSD-96-02	
Marian, M. J.	Tech Specialist			NSD-95-13, NSD-96-13	
McCormack, G	Tech Specialist			NSD-96-02	
Michalski, K.	Tech Specialist			NSD-97-06	
Miller, J.	Tech Specialist			NSD-96-16	
Perkins, B.	Tech Specialist			NSD-96-06	
Perry, S.	Tech Specialist			NSD-95-02	

ASSESSMENT PROCESS REVIEW DOCUMENTATION PERSONNEL QUALIFICATIONS					
Name	**Level**	**Lead Auditor Qualified**	**Comments**	**Audits Performed (L=Lead)**	**Surveillances Performed (L=Lead)**
Petrone, R	Tech Specialist			NSD-96-04	
Phelan, P.	Tech Specialist			NSD-95-04	
Rossman, R.	Tech Specialist			NSD-95-07, VY-96-25	NSD-97-01-01
Salim, Tanvir	Tech Specialist			NSD-96-01	
Scott, M.	Tech Specialist			NSD-95-03	
Sheridan, R.	Tech Specialist		NSD-96-12 marked as "BECO" (ATL) but not shown in ATL Position	NSD-96-12	
Strum, M.	Tech Specialist			NSD-97-10	
Tedeschi, J.	Tech Specialist			NSD-95-02	
Tucker, M	Tech Specialist			NSD-96-02	
Vanvolkinburg, S.	Tech Specialist			NSD-95-03	
Volk, M.	Tech Specialist			NSD-95-04	
Weyman, D.	Tech Specialist			NSD-96-12, NSD-96-13	
Yasi, D.	Tech Specialist			NSD-95-07	

ASSESSMENT PROCESS REVIEW DOCUMENTATION AUDITS				
Report Number (see notes	**Date**	**Area/Activities Covered**	**(D)eficiency/ (O)bservation No.**	**Finding/Observation Description**
NSD 95-01 C/A	Jul 17 - Aug 4, 11,1995	Maine Yankee Project Design Process - Verify the implementation of the QA program for design and design change controls, and to assess the adequacy of the program to meet regulatory commitments and internal Yankee Nuclear services Division (YNSD) engineering procedures	SSCA 0624 (D)	Files containing Safeguards material were found to be incorrectly stored on a local PC hard drive and an uncontrolled floppy disk
			SSCA 0625 (D)	Documentation of the analysis of the seismic integrity of a transmitter in EDCR 95-54 was not included in the design change
			SSCA 0626 (D)	In-process changes to quality documents are being improperly made
			SSCA 0627 (D)	Numerous administrative deficiencies were identified in MY Project calculations
			SSCA 0628 (O)	The effects of the impact of no longer having the Steam Generator blowdown function are not discussed in the Safety Evaluation associated with EDCR 94-50
			SSCA 0629 (O)	There is no process for revising calculations that have been superseded to add a reference to the more current calculation
			SSCA 0630 (O)	Individual training files being maintained by both the Maine Yankee Project and the Training Dept. are duplicative and incomplete

ASSESSMENT PROCESS REVIEW DOCUMENTATION AUDITS				
Report Number (see notes	**Date**	**Area/Activities Covered**	**(D)eficiency/ (O)bservation No.**	**Finding/Observation Description**
NSD 95-02 C/A	May 15-19, 22-23, Jun 2 1995	Vermont Yankee Project - Verify that engineering activities performed by the Vermont Yankee Project (VYP) meet procedural requirements established in the YNSD Engineering Manual (WEs); focus on I&C calculations and Engineering Design Change Requests (EDCR)	SSCA 0603 (D)	SI CAD files were not properly stored
			SSCA 0604 (D)	Calculations are not being turned over to RMS within the required timeframes
			SSCA 0605 (D)	No EDR was generated when Calc VYC-298, Rev 8 was revised to correct an engineering deficiency
			SSCA 0606 (D)	Software usage within Calc VYC-1053 did not meet the requirements described in Rev 15 of WE-103
			SSCA 0607 (D)	EDCRs generated did not follow W-100 requirements
			SSCA 0608 (D)	EDCR page numbers were lacking for certain enclosures and sketches
			SSCA 0609 (D)	Changes to QA requirements in EDCR 94-404 should have required a major ECN
			SSCA 0610 (D)	Some of the review comments were not incorporated into an EDCR even though the preparer had signed off that all comments have been incorporated into text and drawings
			SSCA 0611 (O)	Cross-outs in calculations violate WE-103 requirements

ASSESSMENT PROCESS REVIEW DOCUMENTATION AUDITS				
Report Number (see notes	**Date**	**Area/Activities Covered**	**(D)eficiency/ (O)bservation No.**	**Finding/Observation Description**
			SSCA 0612 (O)	EDCR 94-402 identified that temperature logging of a condensate pump discharge line will be added as an EDCR open item. It wasn't
			SSCA 0613 (O)	TAG 14, Rev 6 does not provide sufficient detail as to how the various groups within YNSD procedurally control the handling of SI
			SSCA 0614 (O)	Procedural enhancements to the WEs have been provided due to the review of engineering documentation
			SSCA 0615 (O)	Enhancements have been provided to the VYP by the audit technical specialist for consideration (traceability and reusability of computer files)
NSD 95-03	Aug 21-24, 29-30, Oct 27, 1995	Seabrook Project - Verify implementation of the Yankee Atomic Electric Company (YAEC) Operational Quality Assurance (QA) Program for Seabrook Project activities; determine effectiveness of the program in achieving quality and operational objectives	SSCA 0634 (O)	Minor inadequacies with calculations were identified
			SSCA 0635 (O)	Calculations lacked information needed for a reviewer to understand the calculation during an independent review
NSD 95-04	Dec 11-17, 28-29, 1995	Yankee Rowe Project - Review of plant-generated documentation for design and design change controls and to assess the adequacy of the program to meet regulatory commitments and internal Yankee Nuclear Services Division (YNSD) engineering procedures	SSCA 0656 (D)	Calculations do not document reviewers comments and their resolutions on Form WE-103-3

ASSESSMENT PROCESS REVIEW DOCUMENTATION AUDITS				
Report Number (see notes	**Date**	**Area/Activities Covered**	**(D)eficiency/ (O)bservation No.**	**Finding/Observation Description**
			SSCA 0657 (D)	Calculations have noted procedural issued identified
			SSCA 0658 (D)	Supporting calculations did not document fully the computing environment
			SSCA 0659 (D)	Calculation shows a structural weld incorrectly annotated on sketch
			SSCA 0660 (O)	Verified computer code was not documented in calculation
			SSCA 0661 (O)	10CFR50 59 SE within EDCR 95-302 does not sufficiently document engineering judgement
			SSCA 0662 (O)	TAG 14 recommends that a list be maintained for Safeguards Information (SI) repositories. This should be implemented in the YRP
NSD 95-06	Jan 15-24, Feb 6, 1995	Northeast Utilities Project - Verify effectiveness and implementation of the Yankee Operational Quality Assurance Program (YOQAP) as approved by NU for contracted QA Category I design and engineering support activities since Sept 1993	SSCA 0572 (D)	The Department Manager did not indicate concurrence of work scope on the Task Initiation Memo prior to start of work for two tasks
			SSCA 0573 (D)	No WE-103-3 calculation review form signed by the reviewer for a calculation
			SSCA 0574 (D)	Neither the Project Manager nor Engineer Manager initiated a note to file for authorization to work prior to receipt of a purchase order
			SSCA 0575 (O)	Drawing referenced in three calculations are dated after Rev 0 of the calculations were reviewed

ASSESSMENT PROCESS REVIEW DOCUMENTATION AUDITS				
Report Number (see notes	**Date**	**Area/Activities Covered**	**(D)eficiency/ (O)bservation No.**	**Finding/Observation Description**
			SSCA 0576 (O)	P.O. required implementation of Rev 23 of the Yankee Operational QA Program when Rev 24 was in effect
			SSCA 0577 (O)	No documented evaluation of outstanding Software Problem Reports for a WO and calc.
			SSCA 0578 (O)	Work for a PO was incorrectly classified as "Q" under the Yankee QAP when work was actually performed under the NU QAP
			SSCA 0579 (O)	There was a NHY form inserted in W.O. 8[illegible]o Calculation NUC-149
NSD 95-07 C, C/A	May 9-30, 1995	Nuclear Engineering Department - Assess the adequacy of NED procedures used to perform analyses/calculations and to evaluate the effectiveness of their implementation	SSCA 0590 (D)	NED calculations not following WE-103 procedural requirements
			SSCA 0591 (D)	Two-year reviews of EIs were not performed
			SSCA 0592 (D)	Project Manager did not notify plants or plant sponsor personnel of potential engineering deficiencies
			SSCA 0593 (D)	Non-Sponsor and non-affiliate outside work is not being initiated by ESD in accordance with approved procedures
			SSCA 0594 (D)	PO did not contain right-of-access and nonconformance clauses as required
			SSCA 0595 (D)	Transient Analysis Group guidelines do not meet the requirements of TAG 21

ASSESSMENT PROCESS REVIEW DOCUMENTATION AUDITS				
Report Number (see notes	**Date**	**Area/Activities Covered**	**(D)eficiency/ (O)bservation No.**	**Finding/Observation Description**
			SSCA 0596 (O)	Suggested that WE-108 be clarified to provide guidance for documenting the engineering application and/or limitations of a computer code as established by verification and/or validation
			SSCA 0597 (O)	Various SRs used to initiate work at YAEC do not contain the same information, nor are they processed/controlled through similar approval cycles
			SSCA 0598 (O)	WE-205 does not clearly define the application of QA requirements, nor does it provide for QAD review of procurement documents to assure proper designation of QA categories on purchase orders
			SSCA 0599 (O)	EIs describing design input have conflicting requirements regarding the approval of design input
			SSCA 0600 (O)	Several NED computer code verification reports were not on microfilm
			SSCA 0601 (O)	NED group guidelines stamped "Non-Q" are not signed/approved and, in some cases, are not consistent with Technical Administrative Guidelines
NSD 95-07B C/A	Feb 27, Mar 1-3, 30, Apr 13-14, 19, 24, 27, 28, May 10-12, 1995	Engineering Services Department - Verify effectiveness and implementation of the Yankee Operational Quality Assurance Program (YOQAP) approved by outside clients for contracted design and engineering support services	SSCA 0616 (D)	ESD procedure for order entry does not define the scope to which it applies and has not been implemented since March 1994

ASSESSMENT PROCESS REVIEW DOCUMENTATION AUDITS				
Report Number (see notes	**Date**	**Area/Activities Covered**	**(D)eficiency/ (O)bservation No.**	**Finding/Observation Description**
			SSCA 0617 (D)	Five of Seven contracts requiring Yankee Operational QA Program implementation did not comply with ESD procedure for establishment of QA requirements
			SSCA 0618 (D)	GE SBWR purchase order QA requirements were not satisfied, i.e. QA Plan, yearly QA report, and 1994 annual audit
			SSCA 0619 (D)	Corrective action for deficiencies identified in 1993 are inadequate in that the same types of deficiencies were identified in this audit
			SSCA 0620 (O)	ESD files are disorganized with retrievability minimally acceptable
NSD 95-12 AP, C, C/A	Oct 16-25, 27, 1995	Environmental Engineering Department (EED) and Environmental Laboratory - Evaluate effectiveness of the EED programs in controlling safety and non-safety related activities; assess the adequacy of the programs in meeting licensing commitments, regulatory requirements and quality objectives	SSCA 0639 (D)	Control of records is not sufficient to ensure all QA records are properly identified and controlled
			SSCA 0640 (D)	The Engineering Design Change Request and Plant Change Request review process was not fully implemented
NSD 95-13 C, C/A	Mar 20-22, 28-29, 31, 1995	Fuel Management Department - Verify implementation of the QA Program to control safety related and non-safety related engineering activities within the Fuel Management Department	SSCA 0583 (D)	FMD is not maintaining training files
			SSCA 0584 (D)	Five out of seven procedures/guidelines have exceeded their two year review cycle

ASSESSMENT PROCESS REVIEW DOCUMENTATION AUDITS				
Report Number (see notes	**Date**	**Area/Activities Covered**	**(D) Deficiency/ (O) Observation No.**	**Finding/Observation Description**
			SSCA 0585 (O)	FMD procedures/guidelines do not reflect the current company organization
Appendix R Root Cause Analysis Task Force	Aug 28 - Sep 15, 1995	Vermont Yankee - Determine why Vermont Yankee Fire Protection is not in full compliance with the requirements of 10CFR50, Appendix R; why not addressed previous identified non-compliance, why "hot shorts" continue to be unresolved; why previous audits did not identify these	ER-0460 (D)	Control of valves required for Alternate Shutdown, not assured
			ER-0471 (D)	Inadequate design of cold shutdown repair activity
			ER-0484 (D)	RCIC in Alternate Shutdown mode not assured for a fire in RB-3
			ER-0507 (D)	SRV solenoid cables susceptible to single hot short
			ER-0533 (D)	Insufficient time to initiate Alternate Shutdown system prior to core uncovery
			ER-0534 (D)	Use of RCIC no guaranteed for fire in RB-2
NSD 96-01 AP, C, C/A	Oct 7-10, 1996	YNSD Maine Yankee Project Engineering - Verify implementation of design activities on the YNSD Maine Yankee Project following the requirements of 10CFR50, Appendix B (Criteria 1,2,3,5,6,15,16, & 17), YOQAP Section 3, YNSD procedures and ANSI N452.11	SSCA 0752 (D)	YNSD calculations not conforming to the requirements of WE-103
			SSCA 0753 (D)	EDCR 96-36, ECN Rev 0 didn't adequately specify special conditions in open item Section 9.0 and the Maine Yankee CED independent review did not identify this discrepancy

ASSESSMENT PROCESS REVIEW DOCUMENTATION AUDITS				
Report Number (see notes	**Date**	**Area/Activities Covered**	**(D)eficiency/ (O)bservation No.**	**Finding/Observation Description**
			SSCA 0754 (D)	Discrepancies for the review of design documents subject to the requirements of 10CFR50.59
			SSCA 0755 (D)	Maine Yankee Project is not documenting formal self-assessments
			SSCA 0756 (D)	Installation of an additional penetration on the outside of a pump room was not evaluated under the environmental section of the EDCR; may impact HELB analysis
			SSCA 0757 (O)	Inconsistencies with implementation of project procedures
			SSCA 0758 (O)	Training provided to YNSD engineering and support personnel does not meet requirements set forth by INPO
			SSCA 0759 (O)	Recommendations for improving the self-assessment process within the YNSD organization
NSD 96-02 AP, C, C/A	Nov 4-8, 11-13, 1996	Vermont Yankee Design Engineering - Verify implementation of the QA program for design and design change controls at YNSD	ER 96-1074 (D)	Calculation deviated from Industry Standards without documented justification
			ER 96-1075 (D)	Calculation does not follow accepted industry practice for analysis of AC circuits without providing adequate justification for the deviation
			ER 96-1076 (D)	Calculation did not have the calculation/analysis review form completed
			ER 96-1081 (D)	Procedure WE-100 noncompliances

ASSESSMENT PROCESS REVIEW DOCUMENTATION AUDITS				
Report Number (see notes	**Date**	**Area/Activities Covered**	**(D)eficiency/ (O)bservation No.**	**Finding/Observation Description**
			ER 96-1082 (D)	Calculation had noncompliance issues per WE-103
			ER 96-1083 (D)	Several EDRs assigned to Vermont Yankee Design Engineering Group have not been completed since 1995
			ER 96-1084 (D)	Calculation had noncompliance issues per WE-103
			ER 96-1085 (D)	Calculation lacked certain accuracies and completeness
			ER 96-1086 (D)	Calculation contained an incorrect assumption
			ER 96-1087 (D)	Vermont Yankee calculations sampled not approved since 1992 and certain approved calculations have not been turned over to RMS in an expeditious manner
			ER 96-1088 (D)	Self-assessment performed by the Vermont Yankee design Engineering group did not meet TAG No. 22 requirements
			ER 96-1089 (D)	EDCR 95-404 indicated open items in the text of the EDCR, yet Section 6, "Open Items Section" did not list any open items
			ER 96-1090 (D)	Management attention is warranted due to the WE-100 and 103 procedural noncompliance issues identified during the audit of the Vermont Yankee Design Engineering Group audit, NSD 96-02
NSD 96-04 AP, C, C/A	Dec 16-20, 1996	Yankee Rowe Project - Verify implementation of the QA program for design and design change controls at YNSD	SSCA 0808 (D)	Inconsistencies identified relative to reviews of Engineering Design Change No, 96-303

ASSESSMENT PROCESS REVIEW DOCUMENTATION AUDITS				
Report Number (see notes	**Date**	**Area/Activities Covered**	**(D)eficiency/ (O)bservation No.**	**Finding/Observation Description**
			SSCA 0809 (D)	YRC-1066, Rev 1 and YRC-1088, Rev 1 both use Stardyne 4.10, but Stardyne has never been verified in accordance with WE-108
			SSCA 0810 (D)	Finding SSCA 0658was closed by QAD based on [im]proper completion of corrective actions, specifically the YRC-1089 verification of Stardyne 4.10 included improper reference to the use of a WE-108 verified code that did not exist
			SSCA 0811 (O)	Preventative action required by EDR 96-32 was due on 10/25/96, however the EDR is still open and should be identified as an open item in EDCR 96-303
			SSCA 0812 (O)	There is no index for Safeguards Information repositories within YRP
			SSCA 0813 (O)	Calculation and accompanying sketch inconsistencies
			SSCA 0814 (O)	Five recommendations pertaining to Self-assessment are offered for management consideration
NSD 96-06 AP, C/A	Sep 9-13, 30, 1996	Engineering Services Department, Plant Support Department, and Purchasing Department - Verify adequacy and effectiveness of performing safety related activities in accordance with the Yankee Operational Quality Assurance Program (YOQAP)	SSCA 0734 (D)	Original contract documents are not being transmitted to RMS for storage until job closeout
			SSCA 0735 (D)	Attachment to drawing not properly completed
			SSCA 0736 (O)	Process for closing out contracts and/or purchase orders was not delineated in ESD Procedure ESD-TEMP-01

ASSESSMENT PROCESS REVIEW DOCUMENTATION AUDITS				
Report Number (see notes	**Date**	**Area/Activities Covered**	**(D)eficiency/ (O)bservation No.**	**Finding/Observation Description**
			SSCA 0737 (O)	The Security and Building Services Dept. did not have in process in-place to ensure that all individuals requiring TAG 14 training are complying with the 30 day "read and sign" requirement
			SSCA 0738 (O)	Self-assessment performed on NDE qualification records was not documented
			SSCA 0739 (O)	Original Vermont Yankee drawings stored in the Drafting Dept. had conflicting revision numbers on the same drawing
			SSCA 0743 (O)	Individuals initiating supply requisitions did not appear to be aware of the requirement of WE-205 and QAD 96-039
NSD 96-07 AP, C/A	May 13-24, 1996	Nuclear Engineering Department - Verify that established programs and procedures were being effectively implemented in the areas of Vermont Yankee (VY) Reactor Stability, 10CFR50 Appendix R, and Loss of Coolant Accident (LOCA) Analysis by the Nuclear Engineering Department	SSCA 0709 (O)	Calculations were not in accordance with the nonmandatory supplemental guidelines
			SSCA 0710 (O)	Calculations supporting memoranda forwarded to plants were not properly reconciled
NSD 96-08	Apr 22 - Apr 30, 1996	Self-Assessment - Assess the effectiveness and implementation of the Self-Ass Program	SSCA 0707 (O)	Five recommendation to improve TAG-22
			SSCA 0708 (O)	Recommendations to improve company-wide self-assessment process

ASSESSMENT PROCESS REVIEW DOCUMENTATION AUDITS				
Report Number (see notes	**Date**	**Area/Activities Covered**	**(D)eficiency/ (O)bservation No.**	**Finding/Observation Description**
NSD 96-12 AP, C, C/A	Sep 30 - Oct 4, 1996	Environmental Engineering - Evaluate effectiveness of the EED programs in controlling safety and non-safety related activities and meeting Technical Specification commitments, regulatory requirements and quality objectives	SSCA 0746 (D)	All Emergency Planning procedures were not reviewed at least every two years
			SSCA 0747 (D)	ALARA training is not presented to newly hired YNSD engineers
			SSCA 0748 (D)	Safety related calculations are not always performed in accordance with WE-103
			SSCA 0749 (O)	Route the Technical Administrative Guideline revision description to personnel for read and sign
			SSCA 0750 (O)	Improve procedure by including a specific EED document matrix as an attachment to better clarify records maintenance requirements
			SSCA 0751 (O)	Recommendations to enhance the Engineering Guidelines
NSD 96-13	May 25-29, Apr 1, 1996	Fuel Management Department (FMD) - Verify implementation of the Yankee Atomic Electric Company (YAEC) Operational Quality Assurance (QA) Program by the Fuel Management Department (FMD); evaluated the effectiveness of FMD program for the control of fuel reloads	SSCA 0681 (D)	Procedures/Guidelines lacked the name of the preparer and in one case, lacked the signature documenting the independent review
			SSCA 0682 (D)	Two procedures exceeded their two year review cycle
			SSCA 0683 (D)	Control of Proprietary and Confidential Information is not maintained in accordance with TAG 16 requirements

ASSESSMENT PROCESS REVIEW DOCUMENTATION AUDITS				
Report Number (see notes	**Date**	**Area/Activities Covered**	**(D)eficiency/ (O)bservation No.**	**Finding/Observation Description**
			SSCA 0684 (D)	Copies of Special Nuclear Materials (SNMs) Inventories scheduled to be performed at Vermont Yankee in 1995 are not maintained on file with the Central Accountability Office
			SSCA 0685 (D)	EDR procedure fails to ensure that EDRs are treated as QA records
			SSCA 0686 (D)	Indoctrination of Personnel and training Records are not maintained in accordance with WE requirements
			SSCA 0687 (D)	Self-assessment activities performed by the FMD are not meeting management expectations outlined in TAG 22
			SSCA 0688 (O)	Administrative controls for department procedure revisions should be enhanced
			SSCA 0689 (O)	Incorporate the Fuel Design Qualifications Program into procedural guidelines for Fuel Management Dept. Procedure FMDP-02 to enhance procedural guidance for Engineering Design Reviews
NSD 96-15	Dec 29, 1996 - Jan 12, Feb 26, 1997	YAEC NSD Nuclear Engineering Department (NED), Engineering to Support Maine Yankee Startup/Cycle 15 - Document results of the YAEC QAD overview of the NED engineering activities required n response to allegations concerning LOCA analysis for the Maine Yankee plant	VSFR 96-005-01 (D)	An uncertainty factor used by ABB-CE for small break analysis was not in accordance with YAEC provided documentation.

ASSESSMENT PROCESS REVIEW DOCUMENTATION AUDITS				
Report Number (see notes	**Date**	**Area/Activities Covered**	**(D)eficiency/ (O)bservation No.**	**Finding/Observation Description**
NSD 96-16	Apr 8-9, 1996	YAEC NSD, Nuclear Engineering Department and FROSSTEY-2 Computer Software - Verify Nuclear Engineering Dept.'s (NED) use of the FROSSTEY-2 computer code to generate design input with Westinghouse (W) fuel data and to provide results of the analysis to Siemens	SSCA 0702 (O)	Modify WE-108 to assure that code changes affecting licensing documents are communicated to the appropriate regulatory agencies
			SSCA 0703 (O)	Document the time line for the effectivity dates of codes, the date of changes, and the date of resolution of error reports in relation to the transmittal of design input between YAEC and SPC
			SSCA 0704 (O)	Suggest that the verification of hotrod.f computer code within MY-1707 include more than one test case
			SSCA 0705 (O)	Suggest that Maine Yankee and Yankee define the "evaluation model" for SBLOCA at Maine Yankee per 10CFR50.46
			SSCA 0706 (O)	Recommend that the basis and source of several assumptions and certain results within Calculation MYC-1707 be stated either in the calculation or in a letter from SPC
NSD 96-17	Sep 1, 1995 - Aug 31, 1996	YNSD Functional Area Assessment - Review Yankee Nuclear Services Division (YNSD) performance and determine the level of future oversight; review Functional Area Assessments: Administrative Services Department, Regulatory & Industry Affairs, and Nuclear Engineering	no.	

ASSESSMENT PROCESS REVIEW DOCUMENTATION AUDITS				
Report Number (see notes	**Date**	**Area/Activities Covered**	**(D)eficiency/ (O)bservation No.**	**Finding/Observation Description**
VY-96-25	Apr 22 - May 10,16, 1996	Environmental Qualification (EQ) - Verify that an appropriate design basis has been established and maintained for the VY EQ Program	ER 96-0277 (D)	The VY EQ Document Revision Instruction (EQDI) process allows for EQ Program revisions without YNSD review and approval
			ER 96-0292 (D)	The VY I&C Group's Service/Qualified Life Tracking Report does not include the replacement requirements for the Technipower power supplies as documented in QRD 32.1
			ER 96-0293 (D)	A complete set of calculations was not found to demonstrate the adequacy of all temperature setpoints
			ER 96-0294 (D)	Inconsistent results for the Reactor Building I(RB) post-LOCA one-year integrated gamma air dose have been identified
			ER 96-0295 (D)	Appendices to calculations have been generated, reviewed, and approved, not microfilmed
NSD 97-06	Jan 13-17, 20-23, 1997	Fitness For Duty (FFD) Access Authorization (AA) Procurement - Verify adequacy and effectiveness of the FFD/AA and Procurement Programs implemented by YNSD	SSCA 0817 (D)	POs issued without certain requirements
			SSCA 0818 (D)	Certain individuals have not met the requirements for CBOP training
			SSCA 0819 (D)	FFD collector is not asking the client to list all prescribed over-the-counter medications on the chain of custody form
			SSCA 0820 (D)	Collection site personnel accept less than 60ml of urine

ASSESSMENT PROCESS REVIEW DOCUMENTATION AUDITS				
Report Number (see notes	**Date**	**Area/Activities Covered**	**(D)eficiency/ (O)bservation No.**	**Finding/Observation Description**
			SSCA 0821 (D)	Permanent Record Book and Collection Site Procedure No. 12.4 asks for initials of collector only and not the signature
			SSCA 0822 (D)	Unauthorized personnel were in the collection area
			SSCA 0823 (D)	Performance data was being submitted to the NRC greater than 60 days
			SSCA 0824 (O)	Smith Kline security containers were not appropriately secured to the collection building
			SSCA 0825 (O)	The random alcohol testing process at YNSD is implemented with only one evidential grade breathalyzer
			SSCA 0826 (O)	Routine and periodic self-assessments are not being performed, tracked and trended by personnel in the Administrative Services Department including FFD/AA and Purchasing functions
			SSCA 0827 (O)	WE-205, "Procurement," does not identify the criteria for determining the safety classification of items and services
			SSCA 0828 (O)	Enhancements to the FFD/AA Policy and Procedures for consideration
NSD 97-10 AP, C	Nov 4-7, 1997	Environmental Laboratory - Effectiveness of implementing the E-Lab QA/QC Manual and supporting procedures	CR 97-0180 (D)	Approximately 75 E-Lab procedures which have not been reviewed within the two-year period
			CR 97-0181 (D)	Superseded procedure found in use

ASSESSMENT PROCESS REVIEW DOCUMENTATION AUDITS				
Report Number (see notes	**Date**	**Area/Activities Covered**	**(D)eficiency/ (O)bservation No.**	**Finding/Observation Description**
			CR 97-0182 (D)	Training folder missing records for annual Chemical Hygiene and Laboratory Safety Training
			CR 97-0183 (D)	Two interim procedure changes for a procedure exceeded the specified due date
NSD 97-13	Jun 1,1996 - May 31, 1997	YNSD Functional Area Assessment - Review Yankee Nuclear Services Division (YNSD) performance and determine the level of future oversight; review Functional Area Assessments: Nuclear Engineering Department, Environmental Engineering Department, and Fuel Management Department	none	
NSD 97-17 AP, C	Jun 25 - Aug 25, 1997	YNSD General Engineering Support Services - Assessment of implementation of YNSD procedures. Evaluated the Nuclear Engineering, Fuels Management, Environmental Engineering, Plant Support and Information Resources for activities associated with Criteria I	CR 97-89 (D)	EED administrative personnel incorrectly identified Reg calculation as a Non-QA record
			CR 97-90 (D)	Radiological Engineering Group Procedures were not being reviewed at least once every two years
			CR 97-91 (D)	There is little formal training for personnel as to the programmatic requirements of the WE-103. Revisions not always read by staff
			CR 97-92 (D)	Incorrect software code cited in safety-related calculation SBC-820
			CR 97-93 (D)	No written documentation to support interface requirements called for in the WE-002

ASSESSMENT PROCESS REVIEW DOCUMENTATION AUDITS				
Report Number (see notes	**Date**	**Area/Activities Covered**	**(D)eficiency/ (O)bservation No.**	**Finding/Observation Description**
			CR 97-099 (D)	Action items identified during NED self-assessments not tracked on the corporate self-assessment tracking system; and completion dates are being extended by NED without concurrence by appropriate management
			CR 97-100 (D)	Calculation SBC-819 Rev 0 is missing Figure 13.2.8 which contains the "keff" values used for processing computer code "shtdwn6.f"
			CR 97-101 (D)	Verification of computer code EIM2 is not clearly documented in calculation SBC-765 Rev 1
			CR 97-102 (D)	Verification of computer code AOCOMP is not clearly documented in calculation SBC-826 Rev 0
			CR 97-103 (D)	Calculation SBC-822 Rev 0 needs to be clarified further to explain why 80 fresh assemblies (CYCLE 6A) was used in the analysis and applicable for CYCLE 6C instead of 84 fresh assemblies (documentation problem)
			CR 97-104 (D)	Calculation SBC-788 Rev 0 and SBC Rev 0 (both applicable to CYCLE 6C) need to be clarified for use of CYCLE 6B core models rather than CYCLE 6 core models
			CR 97-105 (D)	The Approver dates (and in some cases the Preparer and Reviewer dates) indicated on several calculation cover sheets occur after the date indicated in the standard memo transittal of the calculation results

ASSESSMENT PROCESS REVIEW DOCUMENTATION AUDITS				
Report Number (see notes	**Date**	**Area/Activities Covered**	**(D)eficiency/ (O)bservation No.**	**Finding/Observation Description**
			CR 97-106 (D)	SIMULATE-3, Version s3ss50102t was used in Seabrook calculation SBC-819 although it had not been validated for use with the Seabrook model
			CR 97-0107 (D)	Lack of Inter or Intra-departmental end-user notification of subsequent revisions to critical input parameters
			CR 97-0108 (D)	Assessment of the potential impact of computer code the errors on analyses not completed within 30 days
			CR 97-0109 (D)	Lack of training, lack of consistency in planning and documenting, lack of understanding the expectations for performing, and inadequate procedure for performing self-assessments
			CR 97-0110 (D)	Corrective actions developed in response to SSCA 0747 and documented as complete were not completed
			CR 97-0111 (D)	QA records related to the control and accountability of Special Nuclear Material (SNM) training records, EDRs and EDNs are not maintained in accordance with regulatory and YARC program requirements
			CR 97-0112 (D)	FMD self-assessments do not meet YAEC Self-Assessment Program requirements and management expectations
			CR 97-0113 (D)	Implementation of corrective actions within FMD is less than satisfactory

ASSESSMENT PROCESS REVIEW DOCUMENTATION AUDITS				
Report Number (see notes	**Date**	**Area/Activities Covered**	**(D)eficiency/ (O)bservation No.**	**Finding/Observation Description**
			CR 97-0114 (D)	Contrary to requirements, the self-assessment plan for IRD doesn't identify specific areas to assess or schedules to complete them by
			CR 97-0119 (D)	A self-Assessment plan has not been developed by the Plant Support Department as required by the Technical Administrative Guideline No. 22
NSD 97-18	Oct 20 - Nov 21, 1997	YNSD Projects and Support Services - Assessment of implementation of YNSD procedures. Evaluated the Automation Technology, Engineering Services, Human Resources and Development Departments, and the Northeast Utilities, Seabrook, Decommissioning and Boston Edison projects	CR 97-0187 (D)	ATD's AI tracking system does not accurately reflect current due dates
			CR 97-0188 (D)	Checklist Form WE-003-1, "Indoctrination of Personnel," for a staff member was inaccurately completed
			CR 97-0189 (D)	ATD self-assessment activities do not meet the requirements delineated in Technical Administrative Guideline No. 22
			CR 97-0190 (D)	The Functional Area Assessment (FAA) Report prepared by the Automation Technology Department does not include all attributes required by Technical Administrative Guideline No.22
			CR 97-0191 (D)	Automation Technology department procedures are not being controlled and maintained as required by Technical Administrative Guideline No. 21

ASSESSMENT PROCESS REVIEW DOCUMENTATION AUDITS				
Report Number (see notes	**Date**	**Area/Activities Covered**	**(D)eficiency/ (O)bservation No.**	**Finding/Observation Description**
			CR 97-0192 (D)	Northeast Utilities Project Department self-assessment activities do not meet the requirements delineated in Technical Administrative Guideline No.22
			CR 97-0193 (D)	Engineering Services Department did not complete any self-assessment during the last year. Corrective actions within the ESD were determined to be unsatisfactory since self-assessment corrective actions from previous findings were overdue
			CR 97-0194 (D)	Inconsistencies in how Form WE-003-01 was filled out and also in the required training
			CR 97-0195 (D)	One of the corrective actions identified in Part 1F of CR 97-0019 was modified without review committee approval
			CR 97-0196 (D)	HRD self-assessment activities do not meet the requirements delineated in TAG No.22 and manager's expectations
			CR 97-0197 (D)	The Functional Area Assessment (FAA) Report prepared by HRD does not include all the attributes required by TAG No. 22

Notes:

AP = Audit Plan reviewed

C = Checklist reviewed

C/A = Deficiencies reviewed for proper corrective action and closure (as applicable)

ASSESSMENT PROCESS REVIEW DOCUMENTATION SURVEILLANCES				
Report Number (see note below)	**Date**	**Activities Covered**	**Finding No.**	**Finding Description**
NSD 96-03-01	Nov 29 - Dec 7, 1996	Northeast Utilities Project - Design Engineering Activities	N/A	
NSD 97-01-01 C/A	Jan 17-19, 1997	Maine Yankee Project - Review of USNRC G.L. 96-01 Activities	SSCA 0815	Observation - Testing Package revision & review concerns
C/A			SSCA 0816	Observation - Project clarification concerns
NSD 97-01-02 C/A	Mar 25-28, 1997	Maine Yankee Project - Design Activities	SSCA 0831	Calculation deficiencies
NSD 97-01-03	May 8-12, 1997	Maine Yankee Batch V Design Interface Control	N/A	
NSD 97-02-01	35450	Vermont Yankee Project - Design and Self Assessment Activities		4-UNSATs - Self-Assessment (references ER-1088)
NSD 97-02-02	35474	Vermont Yankee Project - Organizational Interfaces and Corrective Actions		1-UNSAT - Corrective Action
NSD 97-02-03	35659	Vermont Yankee Project - SI Review of I&C Record Turnover VY Design Engineering Group	N/A	
NSD 97-02-04	Sep 16-29, 1997	Vermont Yankee Cycle 20 Reload	N/A	
NSD 97-03-01	35472	Northeast Utilities Project - Design Activities	N/A	
NSD 97-04-01	Apr 28 - May 2, 1997	Yankee Rowe Project - Design Activities/Software and Error Reports	N/A	
NSD 97-04-02	Nov 18-20, 1997	Yankee Rowe Final Status Survey (FSS) Data Base Management System (DBMS)	N/A	
NSD 97-06-01	Jun 9-11, 1997	YNSD - Fitness For Duty (FFD)/Access Authorization (AA) Procedures	N/A	
NSD 97-06-02	Sep 9-12, 1997	YNSD - FFD/AA Documentation Folder Review	CR-97-0141	Background investigations did not meet criteria for unescorted access
NSD 97-06-03	Oct 20-21,1997	YNSD - Fitness For Duty (FFD) Record Review	CR 97-0175	Individuals exceed 15-month refresher training
NSD 97-07-01	Jan 20-22, 1997	NED-RELAP5YA Self Assessment	N/A	
NSD 97-07-02	May 6-7, 21-22, 1997	Maine Yankee Batch V Design Interface Control	N/A	

ASSESSMENT PROCESS REVIEW DOCUMENTATION SURVEILLANCES				
Report Number (see note below)	**Date**	**Activities Covered**	**Finding No.**	**Finding Description**
NSD 97-07-03	Oct 6,1997	NED Document Control and Review of Calculations	CR-97-0165	Inaccurate Records Log for calcs
NSD 97-08-01 C/A	Feb 24,1997	Engineering Services - Self Assessment	SSCA 0829 closed per CR 97-0193	No Self-Assessments have been performed
NSD 97-09-01	Feb 24-25,1997	Plant Support Department (PSD) - Document Control/Record	N/A	
NSD 97-11-01	35439	YAEC - Self Assessment - Environmental Engineering Department	N/A	
NSD 97-12-01	Apr 23-24, 1997	Maine Yankee Batch V Design Interface Control	N/A	
NSD 97-13-01	Mar 27 - Apr 24, 1997	YNSD - Corrective Action - SSCAs & EDRs		2-UNSATs - Corrective Action
NSD 97-13-02	35691	Corrective Action - Condition Reports		1-UNSAT - Corrective Action
NSD 97-14-01	35537	QAD Self-Assessment	N/A	
NSD 97-15-01	35464	Generic Concern With YNSD for Their Control and Application of Calculation revisions and Associated	N/A	
Note:				
C/A = Deficiencies reviewed for proper corrective action and closure (as applicable)				

DAVID ALAN WALKER

DUKE ENGINEERING & SERVICES, INC.

SUMMARY

Mr. Walker has over twenty-five years of Quality Assurance (Compliance) experience in the manufacturing, construction, and operation phases of commercial nuclear power projects. Sixteen years with management/consulting responsibilities and eight years as a Quality Assurance (Compliance) Manager.

EDUCATION/TRAINING

B.S., Business Management, University of North Carolina at Wilmington (1975)
Lead Auditor per ANSI N45.2.23 (since 1977)
Graduate of Crosby Quality Management College (1990)
Advanced Leadership Development Program (ABB Impell 1993)

EXPERIENCE

Senior Technical Specialist — 08/96-Present
Duke Engineering & Services

QA responsibilities for the Nuclear Group's Northeastern Region offices and the Federal Group. Responsibilities include: QA reviews and approvals, internal QA audits and surveillance, supplier audits, interfacing with clients and supporting client audits.

Quality Assurance Manager — 01/88-08/96
VECTRA Technologies, Inc. (formerly ABB Impell), Atlanta, GA

Responsible for developing and ensuring compliance with the VECTRA QA Program for the Nuclear Engineering, Fossil Engineering, Government and Waste businesses throughout the United States. Mr. Walker was responsible for the direction and performance of all QA activities including: training, client audits, internal audits and assessments, supplier audits, computer software verification approvals, reviews, etc.

In addition to his normal responsibilities, Mr. Walker supported the development of the QA Program in ABB Impell's office in Manchester, England, and joint CE/Impell offices in Juno Beach, Florida (FPL) and the Calvert Cliffs Site Office (BG&E). He also performed compliance assessments and consulting for Southern Company Services.

Mr. Walker's most notable accomplishments as Quality Assurance Manager were operating with minimum staff/budget without sacrificing quality to clients and greatly reducing the number of internal audit deficiencies through implementation of a Quality Improvement Process. In addition, Mr. Walker's record of successful NUPIC and client audits was excellent year after year.

David A. Walker

Mr. Walker was previously a Lead Senior Engineer in the QA Section for seven months. In that position, Mr. Walker's primary responsibility was ensuring compliance with the Quality Assurance Program by performing internal/external audits and surveillance. Mr. Walker also participated in a Safety System Functional Inspection at the Savannah River Plant and maintenance assessments of Georgia Power Plant's Hatch and Vogtle during his first months at ABB Impell.

Principal Quality Specialist 1981-1987
Gilbert Commonwealth, Columbia, SC

Before joining ABB Impell, Mr. Walker worked with Gilbert Commonwealth for over six years. His last assignment with Gilbert was for five years at a Southeast Utility (SCE&G) in a staff augmentation position where he was responsible for providing QA Program support to the client's Quality Engineering Group during operation of their V.C. Summer Nuclear Station (including participation in four major outages). His tasks included:

- Reviewing/approving technical requirements and adding quality requirements to safety/ quality-related purchase requisitions for spare/replacement parts and plant support services.
- Working with suppliers in identifying "critical to function" items, establishing dedication methods and developing/improving their overall spare/replacement part programs.
- Developing/approving inspection and test methods to be used in dedicating commercial items for use in nuclear safety-related applications.
- Reviewing/approving Engineering dispositions of nonconformances which occurred at the operating plant.
- Reviewing/approving proposed design modifications at the operating plant.
- Performing surveys/audits of suppliers as a Lead Auditor.

Senior Quality Specialist
Gilbert Commonwealth, Mexico City, Mexico

Mr. Walker's first assignment with Gilbert was as a consultant to the Mexican Government on their Laguna Verde Nuclear Project. QA Program management responsibilities included assisting in the development of quality programs and procedures, performing evaluations and audits of suppliers and contractors to assess compliance of quality programs, and assisting in the training of client personnel in quality assurance functions.

Quality Assurance Audit Supervisor 1979-1981
Brown & Root, Bay City, TX

Responsible for coordination of site audit activities and the supervision of eight site auditors. In this role he developed site audit schedules, selected audit teams, approved audit plans and reports, conducted

audits as a Lead Auditor, maintained audit status records, and kept current on regulations. He also had extensive interface with subcontractors, the client (HL&P) and the Nuclear Regulatory Commission on compliance issues.

Quality Assurance Engineer 1976-1979
Babcock & Wilcox, Lynchburg, VA

Responsible for the evaluation of Quality Assurance programs for suppliers of various nuclear products and the review of QA manuals, deficiency reports, and other correspondence concerning suppliers. He was responsible for scheduling, planning, conducting, reporting, and performing follow-up of supplier, internal and site audits. Other duties at B&W included the review and approval of quality documents, ensuring that contract submittal requirements were met, writing and updating administrative and inspection procedures as needed, recommending corrective actions for nonconformances and working on special projects.

Quality Control Inspector 1971-1976
General Electric Company, Wilmington, NC

Responsibilities included setup and inspection of a wide variety of materials, parts, and assemblies, utilizing all types of standard gauges, instruments special gauges and equipment specified in inspection instructions. Other duties included completing detailed reports concerning all nonconforming material, training and directing subordinate personnel and maintaining auditable records.

David A. Walker

JONATHAN S. GERSON

DUKE ENGINEERING & SERVICES, INC.

Title: Quality Assurance Engineer

Years: 23

SUMMARY

Mr. Gerson has more than 23 years of experience in the nuclear field. He has held positions as Quality Assurance Engineer, Technical Analyst, Quality Control Engineer and Procedure Writer. He is experienced in Quality Assurance (QA), procedure writing, training coordination, Quality Control (QC), audit status tracking, procurement documentation reviews and receipt inspection. Mr. Gerson assisted in the development of planning and integration strategies and performance milestone assessments and was instrumental in the preparation of computer generated visual aids for a variety of DOE presentations at the Richland Office.

EDUCATION/TRAINING

AutoCad/Computer Networks, Columbia Basin College
Statistics - Tarleton State University
Quality Engineering Refresher Course - American Society for Quality Control
Fundamentals of Nondestructive Testing - Metal Engineering Institute
Basic Welding - Oswego County BOCES
Probability & Statistics Graduate Courses, State University of New York
BS, Civil Engineering, Northeastern University, 1971
Associate of Applied Science, Math/Science, Dean Junior College, 1966

PROFESSIONAL AFFILIATIONS/CERTIFICATIONS

Member of the American Society of Quality
Certified Quality Engineer, American Society for Quality Control, 1984

EXPERIENCE

Quality Assurance Engineer — 12/97-Present
Duke Engineering & Services (DE&S)

Responsible for Quality Assurance (QA) of the Western Region of DE&S. Duties include reviewing project plans, design specifications, procurement documentation, and performing audits, surveillances and supplier surveillances.

Quality Assurance Engineer — 09/97-12/97
Duke Engineering & Services

Jonathan S. Gerson

Performed Project Plan reviews and QA audits of various DE&S field offices in preparation for assuming QA responsibility of the Western Region of DE&S.

Quality Assurance Engineer 06/97-08/97
Duke Engineering & Services

Deployed to Lockheed Martin at the Department of Energy (DOE), Idaho Engineering & Environmental Laboratory (INEEL), and designated as the National Spent Nuclear Fuel (NSNF) Training Coordinator responsible for ensuring that all NSNF Quality Assurance personnel nationwide were properly indoctrinated and trained to NQA-1 and DOE/RW-0333P requirements. In addition, reviewed and revised various quality assurance program documents.

Procedure Writer 04/97-05/97
Wastren Technical Services, Inc.

Updated and modified Tank Waste Remediation System (TWRS) Basis for Interim Operation (BIO) Administration Controls and created TWRS Cross-Site Transfer Line Diagrams for the Department of Energy (DOE), Richland Office (RL).

Quality Assurance Engineer/Technical Analyst 11/92-10/96
Vectra Government Services, Inc. (formerly Asea Brown Boveri (ABB) Government Services, Inc.)

At the DOE-RL, assisted in the development of planning and integration strategies and performance milestone assessments. Developed and wrote the Planning & Integration (PID) Statement of Work RL Directive. Taught the DOE-RL "Statement of Work" course. Assisted in the development of the RL Self-Assessment Program. Developed and produced computer generated visual aids for the Office of Planning & Integration's Management Systems training. Provided computer assistance and instruction to users of software applications. Published the Government Support Services Contract employee's newsletter for three years.

Quality Assurance Specialist 08/89-11/92
Stone & Webster Engineering Corporation (SWEC)

Developed, wrote and maintained the DOE-RL Quality Assurance Branch NQA-1 procedures. Performed and tracked quality assurance surveillances on a variety of contractor NQA-1 procedures and compliance requirements including safety, procurement, and storage systems. Software Accountability Manager responsible for controlling all the computer software used by SWEC at RL.

Engineering Technical Specialist 03/89-08/89
Stone & Webster Engineering Corporation

Investigated the adequacy of the SWEC training program of QC personnel performing NQA-1 and 10CFR50, Appendix B, criteria inspections at Florida Power & Light's Turkey Point Power Station. Assisted in creating computer generated graphics for the formal presentation to win the prestigious Deming Prize.

Jonathan S. Gerson

Quality Control Engineer 05/86-03/89
Stone & Webster Engineering Corporation

Wrote Texas Utilities (TU) Electric Engineering Assurance procedures at the Comanche Peak Nuclear Power Station construction site.

Quality Control Engineer 11/81-03/86
Stone & Webster Engineering Corporation

At Niagara Mohawk's Nine Mile Point 2 Nuclear Power Station project, conducted quality assurance surveillances of piping, hangers and valve installation. Prepared and maintained 10CFR50, Appendix B, criteria surveillance checklists, and performed quality assurance verification of subcontractor programs. Created and maintained a departmental information flow network to update surveillances, audits and inspections status.

Quality Control Engineer 10/77-11/81
Stone & Webster Engineering Corporation

Performed 10CFR50, Appendix B criteria and purchase specification specified safety-related vendor documentation reviews. Reviewed subcontractor piping and hanger planning packages. Performed contractor monitoring, receipt inspection, ASME mechanical (piping, hangers, valves, and welds) and electrical inspection.

Quality Assurance Engineer 08/74-10/77
Stone & Webster Engineering Corporation

Assisted in writing the SWEC Quality Assurance ASME III Manual and related procedures. Performed final safety-related vendor documentation review for the North Anna I and II Nuclear Power Station project.

Engineer 10/71-08/74
Stone & Webster Appraisal Corporation

Performed technical data research including field surveillance for the commercial appraisal of buildings and structures for utilities and industry. Initiated and developed computer programs for appraisal analysis.

Jonathan S. Gerson

PHILIP R. HORSMON

DUKE ENGINEERING & SERVICES, INC.

Title: Technical Specialist II

Years: 21

SUMMARY

Mr. Horsmon has more than 21 years of experience in the nuclear field. He has held positions as Technical Specialist, QC Inspection Supervisor, QC Planning Engineer, and Senior QC Inspector. He is experienced in the field of nuclear power plant construction and inspection activities as well as Quality Assurance programmatic requirements related to 10CFR50, Appendix B and NQA-1. Mr. Horsmon has been responsible for as many as twenty nuclear inspectors, written inspection plans, written quality assurance procedures, and is a certified lead auditor. He has also been involved with numerous readiness review teams on the High Level Waste Program for the Department of Energy.

EDUCATION/TRAINING

West Nottingham Academy, 1971
English Major, Elon College (three years)
Business Administration and Marketing, Strayer College (one year)

PROFESSIONAL AFFILIATIONS/CERTIFICATIONS

ANSI Certified Level II Inspector
Quality Assurance Lead Auditor

EXPERIENCE

Technical Specialist 01/97-Present
Duke Engineering & Services

Provides quality engineering for DE&S corporate and satellite offices. Responsible for providing and assisting in all quality related matters for the Advanced Nuclear Group. Current responsibilities include maintaining a PIR tracking data base; performs QA reviews associated with the PIR process; issues monthly QA status reports; maintains ongoing QA commitment tracking data base; issues annual QA Program Status Report; conducts QA source surveillances including scheduling, approval of checklist, reviewing reports and SDRs; schedules and conducts supplier evaluations. As lead auditor, duties include developing audit plan, checklist, audit reports, and followup on corrective actions. Participated in thirteen audits and led seven to date.

Philip R. Horsmon

Technical Specialist II
Duke Engineering & Services

11/95-12/96

Provided quality engineering support to the Zaporozhye Spent Fuel Nuclear Project in Energodar, Ukraine. Performed audits to place suppliers on DE&S approved suppliers list for project related activities. Conducted source surveillance inspections on vendors that fabricated components for the spent fuel storage system. Also responsible for QA support/consulting, QA training, internal audits and surveillances as well as assisting in the development of workplace procedures for DE&S project personnel.

Technical Specialist II
Duke Engineering & Services

05/94-10/95

Provided quality engineering support to the readiness review team for TRW on the High Level Waste Project for DOE. Responsibilities included advising and assuring the team leader that reviews were being conducted in accordance with applicable procedures, tracked open item reports until closure, reviewed all reports for accuracy and completeness, coordinated and reviewed the interim/final report for accuracy and completeness before approval by the Readiness Review Chairperson.

Participated in audits for the Civilian Radioactive Waste Management Program. Responsibilities included writing checklist, verification of program requirements, writing audit reports and followup on corrective actions.

Participated in a surveillance at the Defense Waste Processing Facility (Fitrification Plant) at the Savannah River Site for DOE. Duties included comparing QA programs between the RW Group and EM Group to find where discrepancies were in the two programs and documenting the differences on a checklist.

Served as a proposal manager for the National Environmental Group. Duties included assisting marketing directors to coordinate and assemble information that was critical to RFP requests. Also was a lead project contract administrator. Duties included coordinating and communicating contractual obligations with clients, approving all invoices prior to processing, forecasting billing estimates and advising personnel associated with contract work.

Technical Specialist II
Duke Engineering & Services, Inc.

03/94-04/94

Provided quality engineering support to DE&S. Duties included participating in audits and surveillances (internally and externally) for DE&S, developing marketing/strategic plans for the Quality Assurance Department and resolving quality issues within DE&S.

Philip R. Horsmon

Technical Specialist I
Duke Engineering & Services, Inc.
04/91-02/94

Responsible for writing, coordinating, reviewing, and implementing M&O Quality Administrative Procedures. Provided support to QA audits and management surveillances of system activities, site activities, and system engineering support to DOE Office of Civilian Radioactive Waste Management (OCRWM) as requested. Maintained the Corrective Action Data Base for tracking and statusing of Corrective Action Reports. Provided quality assurance technical expertise to OCRWM Management for the temporary and permanent storage facilities for spent fuel storage.

Inspector A
Duke Power Company
03/88-04/91

Performed inspections underground for the Bad Creek Pump Storage Project. Inspections included verifying that all rebar, embeds, electrical and piping have been installed in accordance with design drawings, specifications and procedures. Also conducted testing of concrete in accordance with design specifications. Responsible for the installation/inspection and test results of the piezometer, slope indicators (inclinometer and sondex) vertical settlement devices and observation wells instrumentation.

Agent
Monumental Insurance Company
02/87-03/88

Responsible for debit/agency to service, sell and collect premiums. Coordinated and resolved problems between clients and company headquarters regarding personal insurance programs. Assisted clients with financial planning for education and retirement.

QC Inspection Supervisor
Stone & Webster Engineering Corp.
10/84-05/86

Supervised twenty inspectors in the Structural Electrical Hangar Department which included raceway, studwelding, drilled-in-anchors, tray hangars and conduit support inspections. Duties included reviewing and closing inspection reports, nonconformance reports, rework control forms, work tracking documents and writing/revising inspection plans.

Philip R. Horsmon

QC Senior Inspector 06/82-10/84
Stone & Webster Engineering Corp.

Responsible for the inspection of rebar, embeds, formwork, etc. prior to placement of concrete in accordance with design drawings, specifications and procedures. Also, conducted inspections on tray hangars and conduit supports in accordance with design drawings for verification of location, size, orientation, material and torque values for high strength bolting.

QC Planning Engineer 05/80-06/82
Stone & Webster Engineering Corp.

Responsible for writing/revising civil inspection plans for inspectors use. Content of plans were taken from design drawings, design changes, specifications and procedures. Updated inspection handbooks and reviewed inspection reports for final acceptance prior to client turnover. Reviewed design changes and nonconformance for 10CFR21-55E. Also assigned design changes for work verification prior to incorporation into design drawing and/or specification.

Senior QC Inspector 01/81-06/81
Stone & Webster Engineering Corp.

Responsible for the inspection of rebar, embeds, formwork, etc. prior to placement of concrete in accordance with design drawings, specifications and procedures. Also responsible for three inspectors conducting preplacement inspections in the civil discipline.

QC Inspector II 05/80-01/81
Stone & Webster Engineering Corp.

Responsible for the inspection of rebar, embeds, formwork, etc. prior to placement of concrete in accordance with design drawings, specifications and procedures. Also conducted other civil/structural inspections as required.

QC Inspector I 02/78-04/80
Brown & Root Inc.

Responsible for the inspection of rebar, embeds, formwork, etc. prior to placement of concrete in accordance with design drawings, specifications and procedures.

QC Inspector I 04/76-01/78
Stone & Webster Engineering Corp.

Responsible for the inspection of rebar, embeds, formwork, etc. prior to placement of concrete in accordance with design drawings, specifications and procedures. Also performed inspections on cadwelds including qualification, location, size, material and acceptability.

Philip R. Horsmon

QC Inspector I
Bechtel Power Corp.

11/75-03/76

Responsible for inspection of concrete placement in accordance with design drawings, specifications and procedures.

Philip R. Horsmon

APPENDIX H

VERMONT YANKEE SAFETY ANALYSIS ASSESSMENT SUMMARY

APPENDIX H

VERMONT YANKEE SAFETY ANALYSIS ASSESSMENT SUMMARY

H.1 INTRODUCTION

Scientech conducted an independent technical assessment of Duke Engineering and Services (DE&S) work in progress or recently completed in support of Vermont Yankee. The assessment was performed by two teams of independent experts in transients and accident analysis team and a containment team. The teams spent approximately 40 staff days on site at DE&S offices in Bolton, MA conducting the technical assessment. During that period DE&S personnel were interviewed and work products and reference materials were examined.

H.2 ASSESSMENT TEAM MEMBERSHIP

Two teams of experts examined work in the areas of transients and accident analysis and containment. The transients and accident experts were Ben Gitnick, P.E., and Dan Prelewicz, P.E., Ph.D. The containment experts were John Atchison and Robert L. Tedesco. These two teams have a combined experience in the nuclear power industry of more than 100 years.

H.3 APPROACH

The specific areas for the independent assessment included:

- Basis for Maintaining Operations (BMOs),
- Key Input to the Reload Analysis, and
- Containment Analysis Supporting the Torus Temperature Limit.

To guide the technical assessments, the checklist in Table H-1 was used in the evaluation of each analysis. While reviewers were free to pursue an independent approach to assess the technical quality of work products, the reviewers completed a checklist for each work product. Interviews were held with the authors and/or reviewers of a number of the work products to obtain information needed for the assessment. The interviews afforded the opportunity to obtain additional input on the experience and knowledge of the work product authors and reviewers.

H.4 CONCLUSIONS

The conclusions of the Scientech review team are summarized below.

The quality and professionalism of the work was found to be very good. No safety-significant problems were identified. Several potential problems were identified in the documentation and traceability of work, particularly for BMOs. Recommendations for improvements were identified regarding follow-up of action items identified in BMOs and the approach used to address 10 CFR 50.46 related issues.

Strengths which were readily apparent as a result of the assessment include:

1. The calculations and engineering analyses reviewed were of a high quality and were at or above the standard for comparable industry calculations. A number of procedural and documentation problems were found, but no significant technical errors were uncovered.

2. The DE&S authors and reviewers demonstrated a very high level of competence and knowledge of the technical areas. Questions were answered accurately and correctly without hesitation. Detailed technical explanations were provided without recourse to reference materials and knowledge of the plant and industry practices was evident. Available references and backup materials requested were provided promptly.

3. No significant technical errors were identified during the assessment of each of the specific areas described in Section H.3, attesting to the care and competence of the staff performing the work.

4. A conservative approach was evident in the work.

Weaknesses and Observations were identified in three areas:

1. Although the basis for all BMOs reviewed was assessed to be appropriate, there is a need to document additional information and references which support the judgments presented for a number of the BMOs. In most cases, the information existed and could be provided by the BMO authors. However, the information was not archived, and therefore it will likely become more difficult to retrieve as time passes. Including the information either in the BMO directly or in an archived form referenced in the BMO would preserve the supporting information for future retrieval.

2. Most of the BMOs list required follow up action items. There were some indications that the timeliness of completion of certain action items could be improved.

3. Statements made in the NRC Demand for Information (DFI) letter of December 19, 1997 may be interpreted to mean that 10CFR50.46 ECCS related concerns should only be addressed using an approved evaluation model. DE&S has addressed 10CFR50.46 concerns in BMO 97-38 based on a calculational estimate, i.e., an undocumented calculation using the NRC-approved VY plant model. While internal procedures (NED-5) were followed, in light of the NRC interpretation of 10CFR50.46, DE&S should back up the estimate with a documented evaluation model calculation.

Some of the work which was reviewed was in-progress. In these cases, comments and suggestions were provided to assist DE&S in delivering a high quality product.

TABLE H.1

Technical Assessment Checklist

Criteria 1 - Technical Adequacy of the Work Product

- Are the purpose and intended results of the analysis clearly stated?
- Were the correct inputs used?
- Are the reference sources identified?
- Did the inputs come from approved sources?
- Was the most recent controlling information available and used?
- Are the inputs consistent with the plant design basis?
- Are the inputs reasonable and consistent with inputs to comparable calculations?
- Are the assumptions of the analysis stated?
- Are the assumptions appropriate and justified?
- Is the technical approach conservative, or were uncertainties in results evaluated, particularly in cases where margins are small?
- Are the formulas, correlations and models used appropriate for this analysis?
- Were formulas, correlations and models referenced to acceptable sources?
- How do the methods used compare to generally accepted industry practice?
- Have any problems encountered with computer code results been clearly identified?
- Are the results reasonable and consistent with results from comparable calculations?

Criteria 2 - Compliance with the Governing DE&S Engineering Procedures and Guidelines

- Which specific DE&S procedure(s) apply to this work?
- Was the information needed for the analysis transmitted to the preparer in a controlled manner?
- If a computer code was used, was the code verified and validated in accordance with WE-103 or WE-108 for this type of analysis?

TABLE H.1
(Continued)

Technical Assessment Checklist

- Were all applicable error reports available to the code user?
- Was the computer code referenced on the calculation cover sheet?
- Did the analysis identify all applicable engineering documents requiring revision/review as a result of the conclusions, and have the revisions been made?
- Was the required independent review performed and properly documented?
- Was the review adequate to assure the validity of the conclusions of the analysis?
- Were the required approvals obtained?

TABLE H.1
(Continued)

Technical Assessment Checklist

Criteria 3 - Compliance with Regulatory Requirements

- What regulatory requirements apply to the work?
- What are the current design bases and assumptions?
- Have the SER limitations for any codes used been clearly identified and addressed?
- Are any limitations on the use of the results clearly identified?
- Did the preparer assure that all regulatory requirements have been met?
- Did this work result in a change in the current design basis documentation?

APPENDIX I

SEABROOK SAFETY ANALYSIS ASSESSMENT SUMMARY

APPENDIX I

SEABROOK SAFETY ANALYSIS ASSESSMENT SUMMARY

I.1 INTRODUCTION

Scientech conducted an independent technical assessment of Duke Engineering and Services (DE&S) analyses provided to support North Atlantic Energy Services Company (NAESCo) Seabrook Station. The assessment was performed by a team of three knowledgeable and experienced Scientech engineers. The team spent approximately 20 staff days on site at DE&S offices in Bolton, MA conducting the technical assessment. During that period DE&S personnel were interviewed and work products and reference materials were examined.

I.2 ASSESSMENT TEAM MEMBERSHIP

The assessment was conducted by Thomas R. Hencey, Donald E. Palmrose, Ph.D., and David G. Vreeland, Ph.D. This team has a combined experience in the nuclear power industry of more than 60 years.

I.3 APPROACH

The primary objective of this effort was to assess the effectiveness of the Duke Engineering & Services Inc. (DE&S) analyses which have been performed to support North Atlantic Energy Services Company (NAESCo) Seabrook Station. The technical assessment focused on selected analyses and their translation into licensing and design documentation, using the guidance provided in NRC Inspection & Enforcement Manual Inspection Procedure 93809, *Safety System Engineering Inspection (SSEI) Inspection Procedure*.

Reviews and assessments were conducted of eight calculations and associated calculation revisions which have been performed in support of the operation of Seabrook Station. The verification activities employed to assess these analysis files were specifically designed to determine:

1. Whether the calculations, analyses, and documentation provided by DE&S establishing the design bases of the plant are accurate and complete, and

2. Whether the processes used by DE&S and NAESCo are effective in maintaining the design bases plant documentation, and assuring that effective communication links are established to translate design bases information into appropriate licensing documents.

The specific assessment questions appearing in Table I-1 were used as a guide in performing the evaluation of each analysis file. Each of the indicated analyses were reviewed against the assessment questions identified. Interviews with DE&S and NAESCo personnel were also conducted to ascertain the answers to the assessment questions.

I.4 CONCLUSIONS

The quality and accuracy of the work and the professionalism displayed by it was found to be good. No safety-significant problems were identified. Strengths of the analyses and the program that were noteworthy to the review team are highlighted below. Some documents that were prepared prior to 1991 could have designated the final analyses and results more clearly to provide for ease of review and reference. While there was no evidence to suggest that this has caused inaccuracies in work or any quality problems, these instances are classified as weaknesses since there is some possibility that this type of documentation could cause an improper reference to occur in the fu. e. Recommendations for improvements were identified that could strengthen the program by building links between analysis files and supplementary, related reference material.

Strengths which were readily apparent as a result of the assessment include:

(1) The DE&S authors and reviewers demonstrated a very high level of competence and knowledge of the technical areas. Questions were answered accurately and correctly without hesitation. Available references and backup materials requested were provided promptly.

(2) Calculations, analyses, and documentation provided by DE&S establishing the design bases of the plant are accurate and complete. No significant technical errors were identified during the assessment, attesting to the care and competence of the staff performing the work. This conclusion demonstrates that Objective 1 of the Safety Analysis Assessment, as summarized in Section I.3, has been met.

(3) Processes used by DE&S and NAESCo are effective in maintaining the design bases plant documentation and assuring that effective communication links (people and process) are established to translate design bases information into appropriate licensing documents. This conclusion demonstrates that Objective 2 of the Safety Analysis Assessment, as summarized in Section I.3, has been met. In addition, the NAESCo 10 CFR 50.59 evaluations reviewed by the team were found to be comprehensive and are ranked among the better facility evaluations in the industry which have been seen by the review team. Analysis support and documentation required for these evaluations is frequently provided by the Engineering staff of DE&S who were the subject of this review.

(4) A conservative approach was applied and, when required by the analysis, proper assumptions were made for each of the analyses. Where appropriate, analyses were verified through the use of parametric studies, as well as through comparisons of the results of analysis calculations with plant testing and operational data. These DE&S practices compare well to generally accepted industry practices.

Weaknesses were identified in the following area:

As stated in the summary above, some documents that were prepared prior to 1991 could have designated the final analyses and results more clearly to provide for ease of review and reference. The two instances identified were:

(1) Two radiological dose calculations performed in 1980 and 1981 were reviewed. They were prepared using Standard Review Plan (SRP) analysis approaches and criteria. One calculation has been expanded through several revisions, adding additional applications and results. The other calculation includes an attachment which compares results to a later NRC Branch Technical Position that amended the SRP. Multiple revisions can be an effective documentation approach for an experienced team, but do not present the results in the clearest fashion.

(2) Documentation of some thermal-hydraulic calculations includes sections describing analysis approaches, data sets, or computer runs that are not used to derive the final analysis and results. It is good practice to retain such information. However, consideration should be given to developing a better system to retain this information. There was no evidence to suggest that this practice caused inaccuracies in work or any quality problems. It is classified as weakness since it creates the possibility that, in the future, an engineer unfamiliar with the calculation could reference the wrong information.

The following observations of the review team are provided to assist DE&S and NAESCo to improve their processes:

(1) The files prepared in the last few years are generally more complete than the older files. The current DE&S QA procedures and engineering practices produce documents that are comparable to the documents produced and used by other organizations. Based on the review of a recent calculation, it is concluded that use of these practices and procedures can yield calculations that are traceable for future reference.

(2) Backup materials and references to support the assumptions and judgments presented in the analysis files completed prior to 1991 were not always included or detailed in the file. In most cases, the materials were provided by the authors during the review. There were no deficiencies in quality identified in this process; yet there is a reliance on experienced staff to retrieve this information. NAESCo should evaluate whether these older files should be updated to current procedures and practices to reduce the time devoted to retrieval.

(3) An evaluation performed to determine the post-LOCA boron concentration requirements to maintain core subcriticality, and the time required to transition to hot leg recirculation to prevent boron precipitation, was judged to be technically correct and accurate. The review team recommended that an analysis be performed to validate assumptions regarding the sensitivity of results to the decay heat and initial boron concentration input values.

TABLE I.1

Analysis Files Technical Assessment Questionnaire

Technical Adequacy of the Analyses

1. Are the purpose and intended results of the analysis clearly stated?
2. Were the correct inputs used?
3. Are the reference sources identified?
4. Did the inputs come from approved sources?
5. Was the most recent controlling information available and used?
6. Are the inputs consistent with the plant design basis?
7. Are the inputs reasonable and consistent with inputs to comparable calculations?
8. Are the assumptions of the analysis stated?
9. Are the assumptions appropriate and justified?
10. Is the technical approach conservative, or were uncertainties in results evaluated, particularly in cases where margins are small?
11. Are the formulas, correlations and models used appropriate for this analysis?
12. Were formulas, correlations and models referenced to acceptable sources?
13. How do the methods used compare to generally accepted industry practice?
14. Have any problems encountered with computer code results been clearly identified?
15. Are the results reasonable and consistent with results from comparable calculations?
16. What regulatory requirements apply to the work?
17. What are the current design bases and assumptions?
18. Have the SER limitations for any codes used been clearly identified and addressed?
19. Are any limitations on the use of the results clearly identified?
20. How did the Preparer assure that all regulatory requirements have been met?

TABLE I.1
(Continued)

Analysis Files Technical Assessmen[illegible] Questionnaire

Effectiveness of DE&S and NAESCo Administrative Controls

1. Was the information needed for the analysis transmitted to the Preparer in a controlled manner?

2. If a computer code was used, was the code verified and validated in accordance with WE-103 or WE-108 for this type of analysis?

3. Were all applicable error reports available to the code user?

4. Was the computer code referenced on the calculation cover sheet?

5. Did this work result in a change in the current design basis documentation?

6. Did the analysis identify all applicable engineering documents requiring revision/review as a result of the conclusions, and have the revisions been made?

7. Was the required independent review performed and properly [illegible]ocumented?

8. Was the review adequate to assure the validity of the conclusions of the analysis?

9. Were the required approvals obtained?

10. If applicable, is the FSAR description of the calculation/analysis consistent with the analysis file?

11. Was any associated 50.59 safety evaluation (if applicable) performed in a rigorous manner?

12. Was the UFSAR change procedure utilized (if applicable)?

APPENDIX J

RESUME OF WILLIAM H. RASIN

WILLIAM H. RASIN

DUKE ENGINEERING & SERVICES, INC.

SUMMARY

Senior Executive with highly diversified experience in the Nuclear Industry including operations, maintenance, training, research, engineering, licensing, government/regulatory affairs and policy making. Well known for leadership and team building skills as well as technical competence and professional integrity. Demonstrated ability to communicate with a wide variety of audiences.

EDUCATION/TRAINING

BS, Nuclear Engineering, University of Virginia
Registered Professional Engineer

EXPERIENCE

Management Consulting 1996-Present
Duke Engineering & Services, Inc.

Providing management consulting services to selected clients.

- Served as Deputy Project Director, Technical Baseline, for the Hanford Spent Nuclear Fuel Project while reorganizing the engineering function, resolving major technical issues and reducing capital cost.

- Led a multi-discipline, multi-national group of experts to review the design bases for the spent nuclear fuel environmental restoration activities at the Hanford site.

- Provided organizational and quality management oversight for the new Hanford contractor.

- Provided senior independent review of nuclear vendor design basis reconstitution projects.

Senior Vice President, Technical/Regulatory Division 1994-1996
Nuclear Energy Institute, Washington, DC

Directed the resources of the Nuclear Industry in resolving generic technical/regulatory issues with the Nuclear Regulatory Commission, Department of Energy, Environmental Protection Agency, the Congress and others.

- Organized and staffed the Division.

- Saved the industry several billions of dollars in regulatory related costs.

- Developed the process by which the Nuclear Industry reaches consensus on problems of national impact.

- Initiated industry-wide efforts to define a new, performance-based approach to safety regulation and achieved Nuclear Regulatory Commission acceptance of this approach.

- Obtained practical regulation providing for the renewal of nuclear plant operating licenses allowing continued operation beyond the original forty-year limitation.

- Served as principle liaison with the Institute of Nuclear Power Operation and the Electric Power Research Institute.

- Served as the industry spokesman with key government officials and Congressional staff on industry technical/regulatory affairs.

Vice President and Director, Technical Division 1987-1994
Nuclear Management and Resources Council, Washington, DC

Directed the resources of the Nuclear Utility Industry in resolving technical and regulatory issues with the Nuclear Regulatory Commission and the Department of Energy.

- Organized and staffed the division as the first director.

- Resolved long-standing safety issues remaining from the Three Mile Island accident.

- Resolved long-standing concerns with the seismic safety design of nuclear plants.

- Initiated industry proposed changes to government regulations and achieved more cost-efficient safety regulation.

- Developed and implemented numerous industry self-initiated programs to resolve real or perceived safety problems.

Section Head, Nuclear Plant Analysis 1977-1987
Duke Power Company, Charlotte, NC

Directed several groups of engineers performing systems and safety analyses for nuclear electric generating plants.

- Organized and staffed the section to provide independent nuclear safety assessment for nuclear power plants.

- Led the development of "in house" capability with probabilistic safety analysis.

- Worked with NASA to interchange safety approaches between the space shuttle program and the commercial nuclear program.

Group Head, Nuclear Engineering

Directed groups of engineers with responsibility for nuclear engineering activities of the Design Engineering Department during the design and construction of several nuclear power plants.

- Testified as lead technical witness at Atomic Safety and Licensing Board proceedings.
- Dispatched to Three Mile Island as team nuclear expert immediately after the accident.

Supervisor, Nuclear Fuels Analysis

Supervised nuclear fuel procurement, contract negotiation and new and spent fuel storage design for new nuclear power plants.

- Managed a $2 billion nuclear fuel budget.
- Designed new and spent fuel storage for four nuclear power plants.

Research Reactor Engineer/Senior Reactor Operator 1971-1977
University of Virginia, Charlottesville, VA

Directed the use of the university research reactors by student and faculty researchers.

- Licensed by the Nuclear Regulatory Commission as a Senior Reactor Operator.
- Constructed, licensed and started up a low power student training reactor.

Crew Training Coordinator, S1W Prototype 1963-1971
United States Navy

Directed the training program for Naval Officers and enlisted operators for nuclear powered vessels.

- Achieved Engineering Watch Officer qualification as an enlisted man.
- Achieved a 100% graduation rate for a troubled operator qualification program.

Nuclear Reactor Operator
United States Navy

Served on the construction and initial operating crew for a fleet ballistic missile submarine.

APPENDIX K

LIST OF ACRONYMS

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LIST OF ACRONYMS

BE	Best-Estimate
BMO	Basis for Maintaining Operation
CAD	Computer Aided Design
CFR	Code of Federal Regulations
CPARs	Core Performance Analysis Reports
CR	Condition Report
CRRC	Condition Report Review Committee
DE&S	Duke Engineering and Services, Inc.
Demand	Demand for Information
DFI	Demand for Information
Duke	Duke Energy Corporation
ECCS	Emergency Core Cooling System
EDR	Engineering Deficiency Report
EPRI	Electric Power Research Institute
ESD	YAEC Engineering Services Department
FAA	Functional Area Assessment
INPO	Institute of Nuclear Power Operators
ISAT	Independent Safety Assessment Team
JCO	Justification for Continued Operation
JQAG	Joint Quality Audit Group
JUMA	Joint Utility Management Audit
LBLOCA	Large Break Loss of Coolant Accident
LOCA	Loss of Coolant Accident
MYAPCo	Maine Yankee Atomic Power Company
MYAPS	Maine Yankee Atomic Power Station
NAESCo	North Atlantic Energy Services Company

BE	Best-Estimate
NED	YAEC Nuclear Engineering Department
NEI	Nuclear Energy Institute
NEPA	National Environmental Policy Act
NRC	Nuclear Regulatory Commission
NSARC	Nuclear Safety Review Committee
NSD	YAEC Nuclear Services Division
OIG	Office of the Inspector General
PCT	Peak Cladding Temperature
PSD	YAEC Plant Support Department
QAPD	Quality Assurance Program Description
QATC	Quality Assurance Training Coordinator
SBLOCA	Small Break Loss-of-Coolant Accident
SER	NRC Safety Evaluation Report
SSCA	Status and Summary of Corrective Action
SSEI	Safety System Engineering Inspection
UCS	Union Of Concerned Scientists
VQA	Vendor Quality Assurance
WANO	World Association of Nuclear Operators
WE	YAEC Engineering Instructions
YAEC	Yankee Atomic Electric Company