PDR
Per D. Mossburg

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

CHAIRMAN

January 31, 1997

Mr. Henry R. Myers
Post Office Box 88
Peaks Island, Maine 04108

Dear Mr. Myers:

I am responding to the letter you sent me on October 25, 1996, in which you question the basis for the U.S. Nuclear Regulatory Commission (NRC) staff's January 3, 1996, "Confirmatory Order Suspending Authority for and Limiting Power Operation and Containment Pressure (Effective Immediately), and Demand for Information" (Order) to the Maine Yankee Atomic Power Station, which allowed operation at 2440 megawatts (thermal) (MWt), considering the plant's nonconformance with Three Mile Island Action Plan Items II.K.3.30 and II.K.3.31.

Your letter states that the NRC letter of October 18, 1996, does not address the fact that the NRC staff appears to have allowed Maine Yankee to operate at 2440 MWt without having followed procedures for allowing the plant to operate when it does not conform with TMI Action Plan Items II.K.3.30 and II.K.3.31. As explained in the Order, the NRC staff's letters to you of June 18 and August 9, 1996, and in my letter of October 18, 1996, the Order was issued for the purpose of ensuring the safe operation of Maine Yankee pending completion of the staff's evaluation of the Maine Yankee emergency core cooling systems (ECCS) and containment design. The Order, the NRC staff's letter of April 10, 1996, and my letters of October 18 and December 5, 1996, explain in detail that the staff appropriately determined that operation at a reduced power level and with a reduced limit on containment internal pressure poses no undue risk to the public health and safety pending completion of the staff's evaluation of these Maine Yankee analyses.

Your letter requests documents showing Commission consideration of the Order issued on January 3, 1996, to Maine Yankee. No documents exist responsive to this request because the discussions between the Commission and the NRC staff regarding the order were conducted orally and were not recorded.

Your letter asks when the Director, Office of Nuclear Reactor Regulation (NRR), explained the basis for the Order of January 3, 1996, and whether it was before issuance of the Order. Your letter also asks whether the Director, NRR, explained the use of his authority provided by Section 50.46(a)(2) of Title 10 of the Code of Federal Regulations (10 CFR 50.46(a)(2)). As explained in the enclosure to the staff's letter of May 16, 1996, the staff's letters of June 18, July 9, and August 12, 1996, and my letters of October 18 and December 5, 1996, the Director, NRR, appropriately issued the Order of January 3, 1996, pursuant to his authority under 10 CFR 50.46(a)(2). That

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authority, although not specifically referenced by the Order, is included within the general authority cited in the Order. Moreover, the Office of the General Counsel (OGC) provided advice and counsel to the NRR staff during the development of the Order and, in OGC's view, the Order is legally sound.

Your letter states that the NRC's letter of October 18, 1996, does not indicate any analysis demonstrating that a 10 percent reduction in the maximum power level under 10 CFR 50.46(a)(2) compensates for the increased risk resulting from the nonconformance with TMI Action Plan Items II.K.3.30 and II.K.3.31. Similarly, you state that there is no analysis to show the effect of reduced safety resulting from the nonconformance with TMI Action Plan Items II.K.3.30 and II.K.3.31 on the 90 percent power level limitation. You ask whether the 90 percent limitation is a net restriction of operation under 10 CFR 50.46(a)(2) or a net relaxation of regulatory requirements. As explained in the NRC staff's letters of June 18, and August 9, 1996, and in my letter of October 18, 1996, the Order did not relax regulatory requirements. The January 3, 1996, Order clearly restricted operations at Maine Yankee, as your letter acknowledges. Moreover, as explained in my letters of October 18 and December 5, 1996, and in the January 3, 1996, Order, operation at a power level of 2440 MWt and with a containment internal pressure of 2 psig poses no undue risk to public health and safety.

Your letter states that the NRC staff says that a basis for the January 3, 1996, Order is that the large-break loss-of-coolant accident (LBLOCA) analysis bounds credible accidents. You ask what analysis the NRC staff has done to develop a position regarding power levels at which the LBLOCA bounds credible design-basis accidents, thereby making TMI Action Plan Items II.K.3.30 and II.K.3.31 superfluous. As explained in the NRC staff's letters of June 18, and August 9, 1996, and in my letter of October 18, 1996, the January 3, 1996, Order did not waive conformance with TMI Action Plan Items II.K.3.30 and II.K.3.31. As explained in the NRC staff's letters of April 10 and May 16, 1996, and in the January 3, 1996, Order, the NRC staff judged that the reduction in power level to 2440 MWt was necessary to account for post-Cycle 4 small-break loss-of-coolant accident (SBLOCA) model uncertainties. As required by the Order, the licensee has submitted its evaluation that the SBLOCA for Maine Yankee, under the operating conditions for Cycle 15 at 2440 MWt, continues to be less limiting than LBLOCAs. The licensee analysis confirmed that there is substantial margin to the criteria specified in 10 CFR 50.46, and that the additional effects of less significant parameters or intermediate break sizes between 0.1 ft^2 and 0.5 ft^2 would be accommodated. The NRC documented the results of its audit of the licensee's calculations in NRC Inspection Report 50-309/96-01, dated April 2, 1996 (enclosed). The NRC staff considers operation at 2440 MWt, using the core operating limit parameters based upon analyses performed for operation at 2700 MWt, acceptable. Furthermore, as explained in my letters of October 18 and December 5, 1996, and in the January 3, 19 5, Order, operation restricted to a maximum power level of 2440 MWt and a containment internal pressure limit of 2 psig poses no undue risk to public health and safety.

Your letter asks whether Maine Yankee is in substantial compliance with NRC requirements, and whether the level of compliance has diminished to the point to which protection of public safety cannot be assured in the manner required by the Atomic Energy Act. As explained in my letters of October 18 and December 5, 1996, and in the January 3, 1996, Order, operation of Maine Yankee at 2440 MWt and with containment internal pressure limited to 2 psig, pending completion of the staff's evaluation of the Maine Yankee ECCS and containment pressure response analyses, poses no undue risk to public health and safety. Issues that arise at the Maine Yankee facility, such as a recent cable separation issue that was the subject of a December 18, 1996 Confirmatory Action Letter, and an offsite power source issue that was the subject of a January 30, 1997 supplement to the Confirmatory Action Letter, will be evaluated for appropriate action and impact on risk to public health and safety.

Thank you for the concerns that you have expressed about the operation of Maine Yankee. I have assigned Mr. John A. Zwolinski, the Deputy Director of the Division of Reactor Projects - I/II in NRR, the responsibility of responding to future correspondence from you. However, I will continue to monitor the staff's actions related to Maine Yankee, including your correspondence.

Sincerely,

Shirley Ann Jackson

Shirley Ann Jackson

Enclosure: NRC Inspection Report 50-309/96-01

U.S. NUCLEAR REGULATORY COMMISSION
REGION I

REPORT NUMBER:	50-309/96-01
DOCKET NUMBER:	50-309
LICENSEE NUMBER:	DRP-36
LICENSEE:	Maine Yankee Atomic Power Company 329 Bath Road Brunswick, Maine 04011
FACILITY:	Maine Yankee Atomic Power Station
INSPECTION DATES:	January 1, to February 10, 1996
INSPECTORS:	J. Yerokun, Senior Resident Inspector W. Olsen, Resident Inspector R. Fuhrmeister, Senior Reactor Inspector D. Mannai, Resident Inspector, Seabrook Station B. Korona, Resident Inspector, Pilgrim Station E. Trottier, Project Manager, NRR S. Brewer, Staff Engineer, NRR S. Sun, Staff Engineer, NRR

APPROVED BY: John F. Rogge 4/2/96
J. F. Rogge, Chief Date
Reactor Projects Branch 8

Scope: Resident inspection and safety assessment of plant activities including operations, maintenance, engineering, and overall plant support.

Supplemental inspection coverage of restart activities.

Verification of plant compliance with NRC confirmatory order, dated January 3, 1996, limiting power operation to 2440 MWt (90.3% reactor power) and containment operating pressure to 2 psig.

Overview: See executive summary.

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EXECUTIVE SUMMARY

Operations

Operators performed well, properly followed procedures and used good communication techniques during plant restart. Operators were knowledgeable of plant conditions and system configurations. An instance was noted, however, involving the control elements and power programmer alarms where operators did not anticipate receiving the alarms during control element assemble (CEA) movement. The outstanding work order on the coil power programmer (CPP) alarms represented less than timely correction of an equipment deficiency.

The establishment of additional licensee management oversight during the reactor startup was considered a strength. Operators were attentive and effectively monitored plant parameters during reactor startup. Daily plant management meetings were properly focused on safety issues. Senior station management demonstrated good safety perspectives by thoroughly questioning the staff regarding the emergent work on safety-related systems, such as with the emergency feedwater pump 2S-A and other important key issues relating to the plant startup.

Shift operations supervisors displayed good command and control during the performance of the reactor startup and frequently monitored control room activities to preclude any distractions to the reactor operators. Test personnel were knowledgeable of test procedures, and performed tests satisfactorily.

Effective plant parameter monitoring and a good questioning attitude were evident when the reactor operator performing the startup questioned reactor engineering personnel regarding minimal source range count changes during the initial portion of control rod withdrawal.

Maintenance

Maintenance activities were conducted safely. For example, efforts at addressing the problem with the leak from one of the reactor coolant pump (RCP) motor bearing oil reservoirs was safe and well controlled. During the pump assembly and test run, good management oversight was noted. However, a weakness was identified when personnel initially added oil to the wrong reservoir. Inadequate communication was attributed to the cause of the error.

The planning and scheduling section was effectively implementing the on-line maintenance risk management program. Plant management actively participated in discussions concerning outage times of safety significant equipment. The overall plant risk factors were calculated and considered on a daily basis. In general, system outage times were minimized when they occurred, and personnel maintained the proper safety focus to minimize the impact on safe plant operation.

Engineering

The licensee used adequate methods in the analysis to support Cycle 15 operation at 2440 MWt. The results of the analysis have shown that (1) core operating limits (COLs), established at 2700 MWt by using methods approved for Maine Yankee and not relying on RELAP5YA, bound the operating conditions at 2440 MWt, up to 4000 MWd/Mt cycle exposure, and (2) the procedures and plant modifications, which originally relied on RELAP5YA, remain valid for the current operating cycle.

Actions taken to ensure reactor power limitation to 2440 MWt and containment operating pressure to less than 2 psig to comply with the confirmatory order issued by the NRC were very thorough.

Maine Yankee demonstrated good capability to identify and resolve complex problems as shown with the resolution of the problems with the emergency feedwater isolation valves.

Safety Assessment/Quality Verification

Plant operations review committee meetings were conducted well with evident focus on safety. For example, the procedural controls for limiting the reactor power to 2440 MWt and the containment internal operating pressure to 2 psig were properly discussed focused on safety.

Plant management provided good support to the implementation of the On-line Maintenance Risk Management Program. Significance of equipment taken out for maintenance was properly discussed at the daily morning meetings so that no unnecessary increase in on-line safety assessment plant condition was presented.

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DETAILS

1.0 OPERATIONS

On January 11, 1996, the reactor was made critical for the first time since the plant was shutdown in January of 1995. Low power physics testing was then initiated in accordance with procedure 11-2, Low Power Physics Testing. Power operations was attained later when reactor power was increased above 2%. The main generator was initially phased onto the grid on January 16, 1996 at 8:45 a.m. Following power reduction to conduct startup tests and correct equipment problems, final phase on was achieved on January 18, 1996. On January 22, 1996, the plant reached 90% reactor power where it remained limited by the NRC confirmatory order (2440 MWt, (90.03%)) dated January 3, 1996.

During plant restart. NRC inspectors maintained a round-the-clock coverage of startup activities from January 10 through January 16, 1996. The inspectors observed Maine Yankee operations department personnel perform a startup of the reactor and the plant in accordance with the procedures listed below. The procedures prescribed the processes for operating the plant from Hot Shutdown to taking the reactor critical (Hot Standby) and going up to 10% Reactor Power. On a daily basis, inspectors verified adequate control room staffing, appropriate access control, adherence to procedures and technical specifications limiting conditions for operation, and operability of protective systems, including emergency power sources. The inspectors also verified operability of selected Engineered Safety Features (ESF) trains and assessed the condition of plant equipment, radiological controls, security and safety. The inspectors monitored the status of control room annunciators and radiation monitors to ascertain that they were being responded to and maintained adequately. The inspectors also reviewed the outstanding yellow tags (temporary modifications) prior to startup to ensure that good configuration was maintained and that equipment and systems were in the safe condition. The inspectors evaluated plant housekeeping and cleanliness to ascertain plant condition and ensure no detrimental effect on plant safety was present.

The inspectors attended the daily plant management meetings and found that the meetings properly focused on safety issues. Senior station management demonstrated good safety perspectives by thoroughly questioning the staff regarding the emergent work on safety-related systems, such as with the emergency feedwater pump 25-A and other important key issues relating to the plant startup.

The inspector reviewed the following procedures:

Procedure No. 1-1, "Plant Heatup," Rev.48
Procedure No. 1-2, "Reactor Startup," Rev. 33
Procedure No. 1-3, "Plant Startup," Rev. 45
Procedure No. 11-2, "Low Power Physics Testing," Rev. 18
Procedure No. 11-3, "Power Escalation Tests," Rev 14

The procedures were found to be technically accurate and proper for the evolutions for which they were intended. The inspectors verified that the appropriate controls necessary for reactor power and containment pressure limitations had been properly incorporated into the procedures.

1.1 Plant Heatup

On January 4, 1996, the inspector observed station operators perform a plant heatup to normal operating temperature and pressure in accordance with station procedure 1-1, Plant Heatup, revision 48. The on-shift personnel closely monitored and plotted the actual temperature and pressure relationship in accordance with Maine Yankee Technical Data Book (TDB) figure 1.2.3.2, Heatup Curve. Plant temperature was monitored with reactor coolant system (RCS) cold leg wide range resistance temperature detectors (RTD's) and core exit thermocouples (CET's). The core region subcooling was maintained such that adequate net positive suction head was available for the reactor coolant pumps during the plant heatup.

On January 5, the inspector observed station Instrument and Controls (I&C) technicians, assisted by a vendor representative, perform hot rod testing in accordance with station procedure 3-6.2.1.19, Rod Drop Time Test and Functional Test, revision 12. The inspector noted that the assigned personnel were knowledgeable of the procedure, and that the required testing was performed satisfactorily. Test personnel promptly discussed identified discrepancies with supervisory personnel before continuing on with the procedure. All identified discrepancies were corrected or a station work order initiated to correct the identified concern. The inspector reviewed the completed procedure and verified that all the required test data was properly recorded and within the acceptance criteria.

1.2 Reactor Startup

The inspector observed operators start up the reactor in accordance with station procedure 1-2, Reactor Startup, revision 33. The inspector reviewed the procedure, held discussions with both operators and operations management representatives, walked down the main control boards and back panels, reviewed operator logs and turnover sheets, and independently verified procedure and plant technical specification compliance.

On January 10, 1996, the inspector observed plant operations personnel perform the prerequisites for starting up the reactor. As was being done with all shifts, prior to the on-coming shift personnel assuming their duties, the acting operations department manager conducted a pre-shift briefing concerning the NRC order that limited Maine Yankee's reactor power to 90% and operating containment pressure to 2 psig. The order requirements were incorporated in station procedures and all operations personnel were trained on the requirements of the order prior to assuming a shift. The procedures that were required to be revised due to this order were discussed in detail during the pre-shift briefings to ensure that all operators were knowledgeable of the order requirements prior to assuming their duties.

On January 11, the inspector observed operations personnel conduct the approach to criticality in accordance with the startup procedure. The reactor startup commenced at 1:30 p.m. and the reactor was determined to be critical at 10:20 p.m. A decision was made by the on-shift personnel to use rod withdrawal during the criticality approach instead of boron dilution for more responsive reactivity control. The inspector found this to be well thought of and indicative of a focused and knowledgeable operations crew. The licensed reactor operator performed the operation in accordance with the procedure in an excellent manner with good caution and attention to detail that ensured a safe reactor startup. The shift operations supervisor displayed very good command and control during the performance of the reactor startup and frequently monitored control room activity to preclude any distractions to the reactor operator. Access to the control room was limited to only those required to be present. An on-shift briefing was also presented by the reactor engineering section head in accordance with procedure 0-6-9, Infrequently Performed Procedures. He reiterated Maine Yankee's station management philosophy (caution and conservatism) when performing these type of tasks. A discussion of the expected positive temperature coefficient vice negative temperature coefficient, early in the fuel cycle with the resultant difference in reactivity control was emphasized to the operators. The chain of command was also reiterated and a discussion of the next activity (Low Power Physics Testing) requirements and expectations ensued. The inspector determined that Maine Yankee properly briefed the on-shift operations and reactor engineering personnel with good safety perspective in evidence. A Maine Yankee quality programs inspector also conducted an independent assessment of the plant startup in the control room.

Effective plant parameter monitoring and a good questioning attitude were evident when the reactor operator performing the startup questioned reactor engineering personnel regarding minimal source range count changes during the early portion of control rod withdrawal. Reactor Engineering indicated that the condition was normal given the extended reactor shutdown.

However, the inspector noted that unexpected coil power programmer (CPP) alarms came in during initial control element assembly (CEA) withdrawal. After initial investigation, operators discovered that an outstanding work order (#94-4203) indicated that the alarm would be received any time a CEA was moved. The inspectors considered this a minor weakness regarding knowledge of plant conditions since operators did not anticipate this alarm nor did they initially understand why the alarm was received. Additionally the inspector considered that the outstanding work order represented less than timely correction of an equipment deficiency. Although work was completed on the rod position indication system during the shutdown, several indication problems remained for at least five CEAs including a load sequencing failure, no upper electrical limit (UEL) indication, dual indication of UEL and intermediate position, and improper rod bottom indication when the bank of rods was withdrawn.

Overall, the inspectors determined that the reactor startup was performed safely and effectively. The inspectors observed that operators properly followed procedures and used good communication techniques. Except for the instance involving the CPP alarms, operators were knowledgeable of plant conditions and system configurations. Roles and responsibilities for the reactor startup were clearly established. The establishment of additional licensee management oversight during the reactor startup was considered a strength. Operators were attentive and effectively monitored plant parameters during reactor startup. Inspectors observed strong command and control and procedure adherence during approach to criticality activities using procedure

1-2, Reactor Startup. The inspectors verified proper calculation of 1/M values by reactor engineering personnel. In addition, reactor operators were alert to nuclear power instrumentation and independently calculated the 1/M values. The criticality approach was by boron dilution and the calculated criticality was at a boron concentration of 1,462.5 ppm. Upon achieving criticality, the inspector observed the following parameters:

RPS Wide Range Log Channels: RPS A, B, C, and D at 5 X 10^{-4} each

CEA positions: Shutdown Groups A, B, and C at 184 each
Control Groups 1, 2, 3, and 4 at 184 each
Control Groups 5A and 5B at 128 and 130 respectively

RCS Boron Concentration: 1,462 ppm

RCS Tave: 530 °F

The difference in actual critical condition from the estimated (1462.5 versus 1462 ppm (.004% delta rho)) was negligible. The inspectors concluded that operators and reactor engineering personnel demonstrated excellent technical capability and conducted startup activities well.

1.3 Low Power Physics Testing

The inspectors observed the performance of low power physics testing (LPPT) in accordance with station procedure 11-2, Low Power Physics Testing, Revision 18, starting early in the morning of January 12. In addition to directly observing the testing, the inspectors reviewed the procedure and held several discussions with Reactor Engineering personnel and involved control room operators. Inspectors independently verified selected data for accuracy, proper calculation and satisfaction of procedure acceptance criteria. The difference between predicted results and actual results were well within the allowed tolerance for critical boron concentration, Moderator Temperature Coefficient (MTC), and control rod worth.

The testing was also observed by a representative of Yankee Atomic Engineering Company in support of Maine Yankee's Reactor Engineering Group. Power level was controlled based upon the indications of the wide range nuclear instruments. The test director worked closely with the engineers and licensed operators to ensure that the prerequisites were met that the procedural guidance and limitations were followed and that the test was conducted safely and satisfactorily.

The inspectors noted that the tests were performed safely in a controlled and deliberate manner. Communications surrounding the numerous reactivity changes required by the low power physics testing were good. Reactor Engineers coordinated effectively with licensed operators during the activity. The inspectors verified that low power physics test acceptance criteria were satisfactorily met. Selected independent CEA worth measurement calculation were performed by the inspector. Reactor Engineering personnel were knowledgeable of the LPPT process. Reactivity computer calibrations were current.

1.4 Power Ascension Tests

Reactor power escalation was commenced early in the morning of January 16. Prior to raising power, the Shift Operating Supervisor briefed the control room operating crew on the expected activities, and on precautions for operating with a positive moderator temperature coefficient. The licensed operators exhibited good communications, command, and control of the plant including the secondary plant equipment alignment and testing related to the power escalation.

Following completion of low power physics tests, power ascension was begun. At about 12% power, on January 16, 1996, the main generator was initially phased on to the grid. At about 18% power, problems with a valve (SCC-T-227) in the main generator cooling gas coolers required a power reduction to below 10% so that the generator could be taken off the grid for repairs to the valve. Later on, power escalation was resumed and following other tests including the turbine overspeed trip test, a final phase on was achieved on January 18, 1996, at 6:55 p.m. On January 22, 1996, the plant attained 90% power (2427 MWt, 8?6 MWe). Nuclear Instrumentation (NI) channel C, Delta T Power was at 90.1% and Channel D Nuclear Power at 90.2%. Boron concentration was 1,096 ppm. All control element assembly (CEA) rod groups were at 184 ± 1 steps except for Group 5A which was at 180.

1.5 Chemistry

The inspector observed proper sampling techniques by a chemistry technician taking steam generator blowdown water samples. The purpose of the water test was to identify the levels of fluorides, sulfates, and chlorides in the steam generators as there are specific limits on the maximum levels that may be present prior to changing plant power level. The technician was knowledgeable of the ion chromatography equipment used and of the steam generators' chemical history. The inspector observed proper preparation of the samples, use of the ion chromatography equipment, and evaluation of the results. The inspector also verified compliance with the once per day sampling frequency required by chemistry procedure 3-7-4-2, Secondary System Chemistry Surveillance.

1.6 Reactor Coolant Pump #1 Oil Leakage Problem

On January 28, 1996, Maine Yankee operations personnel noted that the upper bearing temperature for reactor coolant pump (RCP) number (#) 1 was slowly increasing and that the oil level was decreasing. A decision was made to add oil to the bearing reservoir. However, 3.5 gallons of oil were inadvertently added to RCP #2 upper bearing oil reservoir due to personnel error. After discovery of the error, maintenance personnel subsequently added 4 gallons of oil to RCP #1 upper bearing oil reservoir. With the addition of oil the upper bearing temperature deceased slightly and stabilized at approximately 136° Fahrenheit.

Later, on January 31, 1996, the licensee determined that with the increase in bearing temperature and decrease in oil level, more oil was needed in the upper bearing reservoir for RCP #1. A plan was developed to further investigate the problem including making a loop entry to try to determine the reason and source of the oil leakage, the destination of the oil being lost from the bearing reservoir and the reasonableness and safety of further oil addition.

The inspector attended the briefing for the containment and loop #1 entries. The briefing was very comprehensive and involved upper licensee management. Personnel and equipment safety were properly considered. The plan was to enter and conduct visual inspection of the sight glass for oil level and the areas for any oil splashes or spill. Then personnel would add oil as required and drain the collection tank for oil leakage measurement. There were very detailed task, safety, and radiological work safety briefings. The plant shift supervisor (PSS) was designated the working party leader. He was very aware of task termination and/or plant shutdown criteria as discussed during the briefing. Work orders (WO) #96-563 and 96-563-01, were generated for the work activities. Entry into the loop was to be made under radiological work permit (RWP) #96-102, which was also discussed during the briefing.

The results of the inspection identified that the leak was from one of the motor bearing oil reservoir inspection doors at about 2 drops every 20 seconds. No oil splashes or spills were observed. Personnel added about 9 gallons to the RCP upper reservoir and drained about 11 gallons from the collection tank. The inspector concluded that the licensee's efforts at addressing this problem had been safe and well controlled. However, a weakness was identified when personnel added oil to the wrong reservoir initially. Inadequate communication was attributed to the cause of the error.

2.0 MAINTENANCE

Overall, maintenance and surveillance activities continue to be performed well. The inspectors ascertained that activities were performed safely and in accordance with approved plant procedures.

2.1 Maintenance Observation

The inspectors observed and reviewed selected maintenance activities to assure that the activities were conducted safely; complied with technical specifications and work order (WO) requirements; that required approvals and releases were obtained prior to commencing work; that the work procedures were appropriately detailed and followed; and that equipment was properly tested and returned to service. The inspectors observed portions of the following work activities and noted no significant discrepancy:

- WO 94-04057, SW-24, Replace Bolting
- WO 96-0563, #1 RCP Oil Addition
- WO 96-0736, Troubleshoot and repair of 4KV circuit breaker
- WO 96-0516, Repair of P-38

The inspectors also observed portions of work activities on emergency feedwater pump, P-25A under work order 96-00102-02, Disassemble, inspect and reassemble the P-25A spare rotating assemble. There were good foreign material exclusion (FME) controls in place during the maintenance activities. Exposed ends of pipes were taped to preclude the entrance of foreign material into the system. Good quality control (QC) and vendor support was noted during the reassembly. Mechanics properly followed instructions in the work package and assembly procedure. During the post maintenance testing, a steady oil leak was observed from the pump motor end bearing area. The bearing casing was inspected and the oil level was found to be high. The oil level was restored to the proper level and the subsequent functional and inservice tests run were completed satisfactorily. The inspectors noted while the overall maintenance activity had been conducted well, the excessive oil issue represented a minor weakness since the proper amount of oil should have been added in the first case, thereby precluding the need for a rework. During the pump assembly and test run, good management oversight was noted.

2.2 Surveillance Observations

The inspector observed and reviewed selected surveillance activities to assure that the activities satisfied technical specification requirements; that personnel adhered to administrative and surveillance procedures; that test instruments were calibrated; and that test results satisfied the acceptance criteria and when they did not, that the licensee took appropriate actions In addition to the tests discussed in section 1.0, the inspectors observed portions of the following tests and noted no discrepancy:

- Station Procedure 3-1-2.4, Routine ECCS Testing
- Station Procedure 3-:.2.1.19, Rod Drop Time Test and Functional Test

3.0 ENGINEERING

The engineering department continued to provide good support to the plant and maintained good safety perspective.

3.1 Emergency Feedwater Isolation Valves

During the inspection period Maine Yankee continued to experience problems with the emergency feedwater (EFW) steam generator isolation valves (EFW-A 338,339 & 340). These valves failed surveillance leak testing in January 1996. After they were removed, their seat rings were found to be damaged. The valves had previously been repaired to modify the valve seat ring due to excessive "crush" by re-machining the seat retainer groove and the seat ring to a new ½" radius as directed by the valve vendor. The valves are butterfly valves manufactured by Contromatics corporation. The valves' disks are asymmetrical and have seating surfaces similar to that in ball valves.

The EFW isolation valves are required to isolate emergency feedwater flow to a faulted steam generator in an accident situation involving a steam line break. The valves receive a closure signal when the associated steam generator pressure is at 400 psig and decreasing. When the valves receive a signal to close, the faulted steam generator would be at about 400 psig and the upstream pressure is conservatively assumed to be at pump shutoff head of approximately 1500 psig. This would thereby create a differential pressure of about 1,100 psid across the valve. It appears that being designed for this type of a high differential pressure across the valve, the seats have had an inability to withstand the pressure and/or flow forces developed by the normal dynamic conditions of the valve.

Maine Yankee engineering and maintenance personnel investigated the problems with the valve and after consultation with the valve manufacturer developed modifications to resolve the valve problems. When the modifications were completed on the valves, they were taken to a test laboratory where subsequent testing determined that the modifications were successful. The valves were also tested satisfactory after they were installed in the emergency feedwater system. Test results showed very minimal leakage.

The inspector determined that Maine Yankee demonstrated good capability to identify and resolve a very complex problem. Engineers conducted detailed review of the situation and were very proactive at getting the vendor to test and approve the appropriate modifications on the valves. There were good engineering and maintenance skills shown.

3.2 Post Steam Generator Tube Sleeving Reactor Coolant System Flow Rate

The licensee applied for an amendment to the Technical Specifications on April 14, 1995, to allow the use of the Westinghouse Electric Corporation sleeving process for repairing the Maine Yankee steam generator tubes. The amendment was issued on May 22, 1995. Between June and December 1995, Maine Yankee installed sleeves on the inlet side of all steam generator tubes (each of three steam generators contains approximately 5,700 tubes, for a total of approximately 17,000 sleeves). The sleeves were fabricated from Alloy 690 material, while the parent (existing) steam generator tubes are Alloy 600 material. Three sleeve lengths were used: 12, 20, and 30 inches, distributed within each steam generator as follows:

	12"	20"	30"	total
S/G #1	1,329	3,410	706	5,445
S/G #2	1,963	3,141	387	5,491
S/G #3	1,873	3,089	531	5,493

The remaining tubes in each steam generator were not candidates for sleeving, because of the type or location of the existing defect. Those tubes were plugged.

Each sleeve length presents a slightly different hydraulic resistance to RCS flow, with the longest sleeve (30 in.) presenting the most resistance, and the shortest (12 in.) presenting the least. To assist in determining the combined effect on RCS flow of both sleeves and plugs, a "sleeve-to-plug" ratio is used. Because each of the three sleeve lengths presents a different hydraulic resistance to RCS flow, each sleeve has a different total that represents the same resistance as a plug. At Maine Yankee, the approximate sleeve-to-plug ratios are: 50:1 for 12 in. sleeves; 37:1 for 20 in. sleeves, and 28:1 for 30 in. sleeves. That is, approximately 50 of the 12 in. sleeves present the same hydraulic resistance as one plug, while 37 of the 20 in. sleeves, and 28 of the 30 in. sleeves are equivalent to the flow resistance of one plug. As noted above, each steam generator has a different combination of sleeve lengths and total number of sleeves installed. When the combination of sleeve lengths and number installed in each generator is added to the number of plugs installed in each generator, the total can be referred to as a steam generator's "effective" plugging. (Effective plugging is actual plugs plus equivalent plugs.) At the conclusion of the 1995 steam generator outage, Maine Yankee's approximation of effective plugging for each of its steam generators was:

S/G #1   402 effective plugs

S/G #2   351 effective plugs

S/G #3   350 effective plugs,

This was equivalent to an average of 367 plugs per generator, or approximately 6.5% of available tubes effectively plugged. Maine Yankee's cycle 15 analysis allows a maximum of 1000 effective plugs per steam generator, with a maximum asymmetry between steam generators of 500 effective plugs. An NRC specialist inspector reviewed the licensee's methodology for calculating sleeve-to-plug ratios and found it acceptable.

At the conclusion of the sleeving campaign and before reactor startup, the licensee had predicted a decrease in reactor coolant system (RCS) flow of about 2,800 gallons per minute (gpm) from the cycle 14 operating flow. This decrease in flow was expected because of the predicted pressure drop of about 1.7 psi across the steam generators as a result of the sleeves and plugs installed. When flow is calculated based on the enthalpy rise across the core, a flow decrease of approximately 4,000 gpm results.

Using the newly installed and highly accurate loop flow instruments, the licensee determined that the current Maine Yankee RCS flow rate was approximately 375,000 gallons per minute. The flow uncertainty associated with the new instrumentation was calculated to be ±10,000 gpm, which resulted in a possible minimum flow of 365,000 gpm. The minimum RCS flow required by Technical Specification 2.1.1.d is 360,000 gpm, therefore Maine Yankee was restarted within the requirements of the license for RCS flow.

4.0 PLANT SUPPORT

Plant support activities in the areas of radiological controls, security, and emergency preparedness were conducted safely during this period. The inspectors monitored work practices, and conformance to requirements and procedures.

4.1 Radiological Controls

Inspectors routinely reviewed radiological controls including Organization and Management, external radiation exposure control and contamination control. The inspectors also monitored standard industry radiological work practices, and conformance to radiological control procedures and 10 CFR 20 requirements.

4.2 Security

The inspectors verified that security conditions met regulatory requirements, the requirements of the physical security plan, and complied with approved procedures. The inspectors observed security staffing; protected and vital area barriers; vehicle searches and personnel identification; access control; badging; and to assure that they were in accordance with requirements and that appropriate compensatory measures were used when required.

4.3 Emergency Preparedness

The inspectors verified that the emergency response facilities were well maintained and kept ready for use in emergency situations.

5.0 SAFETY ASSESSMENT/QUALITY VERIFICATION

5.1 Plant Operations Review Committee (PORC) Meeting

On January 7, the inspector attended a combined Plant Operations Review Committee/Nuclear Safety Audit Review Committee meeting at the Maine Yankee corporate office. The purpose of the meeting was to review and comment on the Maine Yankee response to the NRC Order and Demand for Information received concerning the allegation regarding the development and use of RELAP5YA LOCA code. The Maine Yankee Vice President of Engineering and Licensing provided background information regarding the coordinated response to the allegations in the area of Yankee Atomic Electric Company's development and use of RELAP5YA computer code. Also the actions of the Allegation Response Team and the two Independent Review teams were reviewed. The overall conclusion of these efforts was that the current licensing basis was appropriate, however areas for improvement were identified. The company president attended the meeting and advised the committees that their function was to review the information to be submitted to the NRC in response to the demand for information and verify that it was complete and accurate, and he was present only as an interested observer.

Members of the PORC questioned what changes needed to be made to th Containment Weight of Air Monitoring Program to account for the new upper limit on containment pressure. The program provides on-line containment leakage monitoring, and may require modification to reflect the revised pressure limit. PORC members requested that the program be reviewed and any necessary revisions be made and provided for their review.

The PORC met again on January 8, 9, and 10 to review the completed analyses and procedure revisions. On January 8 and 9, 1996, the inspectors attended the meetings during which the procedural controls for limiting the reactor power to 2440 MWt and the containment internal operating pressure to 2 psig were discussed and approved. The inspectors noted that the discussions were very thorough and safety focused and appeared technically sound. The inspectors independently reviewed some of the affected plant procedures and noted no issues not being addressed by the PORC.

5.2 On Line Risk Management

Upon returning to power operations, Maine Yankee commenced a new On-Line Maintenance Risk Management Program. The NRC required Maintenance Rule, which goes into effect July 10, 1996, directs that planned on-line maintenance be evaluated and scheduled to minimize the impact on plant safety. The Maine Yankee program document outlines definitions, defines responsibilities and describes the actual assessment process. Assessment is accomplished by a computer-based model, designed from inputs from the Maine Yankee Probabilistic Risk Assessment (PRA). All Maintenance rule identified risk significant systems are considered as well as significant external events when making a risk determination. The PRA risk achievement worth of each system, sub-system, train or individual component is used as a weighing factor when calculating the overall relative margin of safety for key safety functions and the overall plant. The assessment model is used to develop schedules that minimize risk associated with on-line maintenance and also to assess the impact of equipment failures or other unscheduled activities on plant safety. The planning and scheduling section is responsible for maintaining the On-Line Risk Management Program.

The inspector determined that the planning and scheduling Section was effectively implementing the on-line maintenance as described in the program. Plant Management actively participated in discussions concerning outage times of safety significant equipment. The overall plant risk factors were calculated and considered on a daily basis.

6.0 NRC CONFIRMATORY ORDER DATED JANUARY 3, 1996.

On December 4, 1995, the NRC received an allegation against Yankee Atomic Electric Company (YAEC), acting as agent for the Licensee (Maine Yankee Atomic Power Company, or MYAPCo). In brief, it was alleged that YAEC knowingly performed inadequate analyses to support two license amendments to increase the rated thermal power at which Maine Yankee may operate. It was further alleged that Maine Yankee was cognizant of these inadequate analyses, yet misrepresented them to the NRC in seeking the license amendments, which were granted.

As a result of these allegations, the NRC conducted a technical review and evaluation of the circumstances and records surrounding the applications to increase MY's maximum rated thermal power. This review and evaluation was conducted at YAEC Headquarters in Bolton, Massachusetts, on December 11-14, 1995, by a five-member NRC team. The NRC team was accompanied by two representatives of the State of Maine.

On December 18, 1995, the NRC held a public, transcribed meeting in its headquarters in Rockville, Maryland. This meeting was held to afford Maine Yankee and YAEC the opportunity to further describe the use of computer code RELAP5YA at Maine Yankee, and to present information to the NRC to help the NRC determine if its regulatory requirements related to small-break loss-of-coolant accident (SBLOCA) were satisfied for Maine Yankee. The NRC determined that RELAP5YA, which was proposed for use in the current operating cycle (cycle 15) SBLOCA analyses to demonstrate, in part, compliance with emergency core cooling requirements specified in 10 CFR Section 50.46, had not been applied in a manner conforming to the requirements of 10 CFP Part 50, Appendix K, "ECCS Evaluation Model," nor had RELAP5YA been applied in accordance with the conditions specified in the staff's related safety evaluation dated January 30, 1989.

On December 19, 1995, Maine Yankee notified the NRC staff by telephone that there were other uses of RELAP5YA in applications other than SBLOCA. On December 22, 1995, Maine Yankee submitted a letter documenting their commitment to limit reactor power to 2440 MWt and containment pressure to 2.0 psig. Further, Maine Yankee committed to document the justification for use of operating cycle 15 limits, using methods approved for Maine Yankee and without reliance on RELAP5YA. Finally, Maine Yankee committed to conduct a review to identify other applications of RELAP5YA to be used in cycle 15 and verify that operability, as defined in its Technical Specifications, of affected systems and components is maintained.

On January 3, 1996, the NRC issued a confirmatory order and demand for information to Maine Yankee pertaining to the requirements for restart and return to operation at 2440 MWt, and the requirements for return to operation at the currently-licensed maximum power level of 2700 MWt.

On January 10, 1996, Maine Yankee submitted the information required in Section IX of the NRC's January 3, 1996, order and demand for information. The submittal satisfied the four "restart" requirements of the order.

On January 11, 1996, the Maine Yankee reactor was made critical, and on January 16, 1996, the Maine Yankee generator was connected to the Maine electrical distribution grid.

On January 24-26, 1996, an NRC inspection was conducted at YAEC Headquarters in Bolton, Massachussets. The purpose of this inspection was to review and verify the detailed files and computer analyses supporting the licensee's submittal of January 10, 1996. Specifically, the evaluations performed to justify the following:

(1) use of Cycle 15 operating limits without reliance on RELAP5YA,

(2) the operability determinations associated with all other applications where RELAP5YA is relied on for Cycle 15 operation,

(3) the measures taken to limit reactor operation to a maximum thermal power of 2440 MWt (approximately 90% of 2700 MWt), and

(4) the measures taken to limit containment internal operating pressure to a maximum of 2 psig.

In addition, the team reviewed the reactor coolant system (RCS) flow calculations and measurements to determine the effect of the steam generator repairs performed during the 1995 refueling outage. The inspector's findings in each of the areas are discussed below. The findings with regard to RCS flow are documented in Section 3.0 of this report.

6.1 Description of Justification for Cycle 15 Operation at 2440 MWt

The team inspected and evaluated supporting analyses the licensee use to estimate the effects of model changes and plant modifications on peak cladding temperature (PCT), and the cycle 15 core performance analysis report.

6.1.1 Effects of Model Changes

The licensee based SBLOCA analyses for cycle 15 operation at 2440 MWt on its cycle 4 analyses. The cycle 4 analyses were performed assuming a power level of 2630 MWt and used Combustion Engineering (CE) analysis methods that were approved by the NRC. The CE methods do not involve the use of RELAP5YA. However, since cycle 4, CE has modified its methods and Maine Yankee has implemented plant modifications. Therefore, the staff reviewed the licensee's assessment of the impact of these changes on PCT calculations.

The changes in the CE methods for SBLOCA analyses between cycle 4 and cycle 15 are: (1) a revised SG heat transfer model to account for the effects of condensation in the primary side when the SG tubes drain, and (2) addition of a level swell model to account for variations in drift velocity as a function of pressure and the effects of the power shape on the bubble production rate.

6.1.2 Effects of Plant Modifications

Modifications (including changes for cycle 15 operation) to the Maine Yankee plant resulted in changes in (1) moderator temperature coefficients, (2) power shape and peaking factors, (3) flow resistance effects due to SG plugging, (4) SG heat transfer coefficient, (5) cold leg temperature, (6) fuel rod heating effects, and (7) power level.

The licensee has assessed the effects of plant modifications and CE method changes on the results of SBLOCA analyses. Using the cycle 4 calculated PCT as the baseline, changes in CE methods and plant modifications were individually assessed to determine each incremental change in PCT. These changes were added to or subtracted from the PCT value for cycle 4 to calculate a PCT estimate for cycle 15. The results of this incremental calculation show that the estimated maximum PCT for an SBLOCA is less than 1760 °F. To determine the conservatism of the assessment, the licensee performed a second analysis. The second analysis used the NRC-approved CE method of 1977--with a Maine Yankee plant model representing cycle 15 operating conditions--to perform a direct calculation of PCT at 2440 MWt for a spectrum of break sizes. The results of this second analysis (direct calculation) show that the calculated maximum PCT for an SBLOCA is less than 1620 °F. The results of both analyses confirm that there is substantial margin between the more conservative maximum SBLOCA PCT of 1760 °F and the PCT limit of 2200 °F specified in 10 CFR 50.46, and between the same more conservative maximum SBLOCA PCT of 1760 °F and the LBLOCA maximum PCT of 2171 °F at 2700 MWt for Maine Yankee. Therefore, the analyses provide the basis for concluding that the PCT for an SBLOCA at 2440 MWt is bounded by the PCT for a LBLOCA at 2700 MWt and the LOCA-related operating limits for Cycle 15 are restricted by the LBLOCA limits.

6.1.3 Core Performance Analysis Report (CPAR)

The team inspected the supporting analyses the licensee used to develop the core performance analysis report (revision 2) for Cycle 15 operation at 2440 MWt. The supporting analyses formed the bases for operation to 4000 Mwd/mt cycle exposure, and the assumptions used for the loss of feedwater event analysis.

In the original cycle 15 CPAR, the licensee used NRC-approved methods to perform design basis LBLOCA analyses and non-LOCA transients. Thus, none of the cycle 15 core operating limits (COLs) for operation at 2700 MWt were limited by SBLOCA analyses and therefore the COLs were not defined by RELAP5YA analyses. The licensee also performed core physics analyses for core depletion up to 4000 MWd/mt cycle exposure at 2440 MWt and confirmed that the range of the physics parameters (including core peaking, CEA worth, reactivity defects, and coefficients and kinetics parameters) is bounded by that used in the determination of the original cycle 15 COLs at 2700 MWt. These core physics analyses provide the basis for concluding that the original Cycle 15 COLs remain valid for operation at 2440 MWt, up to 4000 MWd/Mt cycle exposure. The licensee will reevaluate selected physics parameters should it operate beyond 4000 MWd/mt cycle exposure at 2440 MWt.

The team found that cycle 15 CPAR credits the automatic steam generator (SG) blowdown isolation on low SG level for mitigating the consequences of a loss-of-feedwater event. This is a design-basis event (DBE) as analyzed in Chapter 14, Safety Analysis, of the plant's final safety analysis report (FSAR). The SG blowdown isolation system was recently installed and is designed to safety-related requirements, including redundant Class 1E power supplies, separate trains, actuation from the reactor protection system, and redundant valves procured and maintained to Quality Program standards. The licensee relies on administrative procedures to control operability of the SG blowdown isolation system, but proposed to add the blowdown isolation system valves to the Technical Specifications when the issue was raised during the inspection. The current procedures allow 3 days of operation if the system is inoperable, and 14 days if partially disabled (that is, single train flow). The procedure that controls this function is a Class A procedure, which Maine Yankee uses for safety related equipment.

The staff considers the equipment adequate for credit in the FSAR Chapter 14 safety analysis. However, the blowdown system isolation valves are not currently in the plant technical specifications (TS). Subsequent to the inspection, the staff reviewed the Maine Yankee TS and noted that valves credited for similar functions in the FSAR Chapter 14 Safety Analysis were included in the plant TS. The staff has therefore determined that the blowdown system isolation valve closure function should be added to the TS and concurs with the licensee's decision to add these valves to the plant Technical Specifications. The licensee currently is preparing a request to amend its operating license to accomplish this.

6.2 Description of Assessment of RELAP5YA Applications Supporting Maine Yankee Cycle 15 Operation

On February 2, 1996, Maine Yankee submitted its schedule for producing the remaining information required by Section IX of the order. Specifically, Maine Yankee submitted its schedule for providing an SBLOCA analysis that is specific to Maine Yankee for operation at power levels to 2700 MWt, and an integrated analysis of the containment. (The SBLOCA analysis is to be submitted no later than May 1, 1996, and the integrated analysis of containment is to be submitted no later than October 1, 1996.)

6.2.1 Operability Determinations for Other RELAP5YA Applications

As noted above, Maine Yankee committed to conduct a review to identify other applications of RELAP5YA to be used in cycle 15 and verify that operability, as defined in its Technical Specifications, of affected systems and components is maintained.

The licensee initially identified 16 applications of RELAP5YA to be used in supporting cycle 15 operation. The licensee found that six applications are not applicable to cycle 15. These applications are: steam generator blowdown tank analysis, simulator benchmarking, turbine building environmental qualification, emergency drill support (modeling an SBLOCA), reactor pressure vessel temperature for low temperature overpressurization considerations, and the use of ZIRLO fuel cladding. The use of RELAP5YA in five of these six cases was either in support of a cycle or application previous to cycle 15, or to confirm or verify another calculation. In the case of ZIRLO fuel cladding, the Commission issued Amendment No. 155 to Maine Yankee's Facility Operating License on February 29, 1996, in response to Maine Yankee's application dated August 30, 1995. The licensee's cycle 15 core contains no ZIRLO clad fuel, but future cores may, consistent with Amendment No. 155 and the terms and conditions of its Operating License. The inspectors reviewed the documentation of two more RELAP5YA applications that the licensee determined were not applicable to cycle 15. These applications were the simulator benchmark analysis and the turbine building environmental qualification analysis.

The inspectors found that the licensee used RETRAN-02/MOD05 and RELAP5/MOD3 for simulator benchmarking. The events analyzed for the simulator benchmark are consistent with that specified in ANSI/ANS-3.5, "Nuclear Power Plant Simulators for Use in Operator Training and Examination." RETRAN-02/MOD02 is an NRC-approved code to analyze a steam line break (a non-LOCA event) in licensing applications. RETRAN-02/MOD05 is an updated version and retains mathematical schemes and physical models of RETRAN-02/MOD02. RELAP5/MOD3 was developed for the analysis of LOCA events. RELAP5/MOD3 results have been compared with test data, including the LOFT L2-5 test for prediction of non-equilibrium condensation behaviors. The inspectors noted that two additional small-break LOCAs (0.01 and 0.15 ft^2 breaks in the hot leg) were run using RELAP5YA for additional simulator validation. However, these two events are beyond the recommendations of ANSI/ANS-3.5. The staff therefore concludes that the licensee's simulator benchmarks have been performed to industry standards, without relying upon RELAP5YA, and the codes applied are appropriate for the events analyzed.

For the turbine building environmental qualification calculations, inspectors found that the licensee and its contractor used hand-calculated methods to determine mass and energy releases resulting from LOCAs. The resulting mass and energy releases were used as input to RELAP4/MOD5 to calculate the temperature profile in the turbine building. The staff judges this to be an appropriate methodology for turbine building environmental qualification calculations.

The licensee performed formal, documented operability determinations for the 10 remaining applications of RELAP5YA that support operating cycle 15. All 10 operability determinations were performed in accordance with station procedure 1-200-2, Operability Determination, "Ensuring the Functional Capability of a System or Component," Revision 0. Each of these operability determinations was properly documented on Attachment A (sheets 1 and 2) to the procedure, with supporting documentation attached.

The inspectors performed a detailed review of four of these operability determinations as a sample of the adequacy of the 10 remaining operability determinations. These operability determinations were: containment environmental qualification, reactor coolant pump trip study, spurious opening of a power-operated relief valve (in support of the 10 CFR 50 Appendix R fire protection program), and the containment. (Note that the operability determination performed for the reactor containment building is not strictly a RELAP5YA issue. The licensee performed this operability determination because of the December 4, 1995, allegation that the 55 psig containment design pressure could be exceeded during a postulated loss-of-coolant accident if the containment was at its maximum allowed initial pressure of 3.0 psig.) The licensee's use of RELAP5YA is limited to non-design-basis events for calculation of mass and energy releases to address issues such as these. These analyses are less complex than licensing basis LOCA calculations and do not involve PCT calculations. They are not used to meet the 10 CFR 50.46 requirements. The staff's previous assessment of RELAP5YA is sufficient to support the limited applications of RELAP5YA. Furthermore, the results of mass and energy calculations inspected do not show unstable and divergent mathematical solutions, such as the oscillations observed in the PCT calculations identified in the NRC's order of January 3, 1996.

The inspector found these operability determinations properly documented, supported by analysis, references, and attachments, and satisfactorily completed. Each was signed by the originator and two concurring reviewers.

Based on the results of our inspection and assessments, the team finds that the licensee has used adequate methods in the analysis to support Cycle 15 operation at 2440 MWt. The results of the analysis have shown that (1) core operating limits (COLs), established at 2700 MWt by using methods approved for Maine Yankee and not relying on RELAP5YA, bound the operating conditions at 2440 MWt, up to 4000 MWd/Mt cycle exposure, and (2) the procedures and plant modifications, which originally relied on RELAP5YA, remain valid for the current operating cycle. Therefore, the team concludes that Conditions 1 and 2 imposed in the January 3, 1996 NRC Order are adequately addressed for operation at 2440 MWt, up to 4000 MWd/Mt cycle exposure.

6.3 Power limitation to 2440 MWt

The inspectors reviewed the actions taken by the licensee to comply with limiting power operation to 2440 MWt (90.3%). The actions involved procedure changes and personnel briefings but did not involve any physical hardware changes. Procedure 1-4, "Operations at Power," and procedure 1-4-2, "Power Level Control" were revised via temporary procedure change (TPC) to eliminate full power operation. In addition, precautions were added to not exceed 2440 MWt reactor power. Copies of the order limiting reactor power were distributed to all holders of controlled copies of technical specifications with a cover letter directing that the order be placed immediately following the facility operating license. The Reactor Engineering instructions and Daily Plant Status report were revised to show a limit of 2440 MWt for reactor power.

The Assistant Manager of Operations met with each operating crew to explain the order and the measures taken to limit reactor power. On January 8, 1996, the inspector attended a session during which operators were briefed on the issues. The briefing was conducted by the assistant manager of operations and was very comprehensive. All licensed operators were required to be briefed prior to standing a watch with the reactor critical. A summary of the actions taken was placed in the Operating Crew Document Review Book. Earlier on the evening of January 14, it was noted that one of the licensed operators had not been briefed on the confirmatory order and actions taken to limit reactor power and containment pressure. The individual was not allowed to stand watch that shift until he had been provided the required briefing.

The inspectors found Maine Yankee's actions to ensure power limitations to 2440 MWt to be thorough. The inspectors independent review of procedures revealed no discrepancies. The inspectors also reviewed the technical specifications and portions of the final safety analysis report to determine if any other changes or physical modifications would be required to ensure full compliance with the power limitation. None were identified. The inspectors particularly verified that no setpoint change to any of the reactor protection system signals was required.

6.4 Containment internal operating pressure limitation to 2 psig

The inspectors also reviewed the licensee's actions taken to limit the containment operating pressure to 2 psig or less. The changes involved procedure revisions, computer alarm setpoint modifications, instrument re-calibration and personnel briefings.

Procedures 1-2, "Reactor Startup," 3-1-1, "Instrument Surveillance," and 1-12-2, "Containment Leak Monitoring" were revised via temporary procedure change (TPC) to reflect a containment pressure limit of 2 psig rather than 3 psig. Abnormal operating procedures 2-37R, "PanAlarm Response," 2-10, "Loss of Pressure Control/RCS Leak," and 2-30, "Loss of Containment Integrity" were revised via TPC to change the 3 psig limit on containment pressure to 2 psig. Main control board annunciator "Containment Pressure Hi" setpoint was changed from 3 psig to 2 psig. Computer alarm points relating to containment pressure were reset to lower values. Operating crew briefings were conducted as noted above. Copies of the order were sent to all holders of controlled copies of the facility technical specifications as noted above. A summary of the actions taken was placed in the Operating Crew Document Review Book as noted above.

An interpretation was added to the Technical Specification Interpretation Manual to reflect the fact that the limit on containment pressure during routine operation is 2 psig, not the 3 psig listed in the technical specifications.

The inspectors were satisfied that Maine Yankee had taken appropriate actions to ensure that the containment operating pressure would be limited to less than 2 psig.

7.0 ADMINISTRATIVE

7.1 Persons Contacted

During this report period, inspectors conducted interviews and discussions with various licensee personnel, including plant operators, maintenance technicians and the licensee management.

7.2 Summary of Facility Activities

On January 25, 1996, the inspectors briefed the Maine State legislatures' Joint Committee on Utilities and Energy on the scope and status of the NRC's investigation of the issues involving the emergency core cooling system and the containment.

During the inspection period the inspectors conducted backshift inspection on January 2, 3, 4, 5, 8, 9, 17, 25 and 26, and February 5, 6 and 8, and deep backshift inspection on January 7, 17, 21 and 26. Between January 10 and 15, 1996, the inspectors maintained a round the clock coverage of plant startup activities.

7.3 Interface with the State of Maine

Periodically, the resident inspectors and the onsite representative of the State of Maine discussed findings and activities of their corresponding organizations.

7.4 Exit Meeting

Inspectors periodically held meetings with senior facility management to discuss the inspection scope and findings. At the conclusion of the inspection, the inspectors also presented a summary of findings for the report period.