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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 77

FACILITY OPERATING LICENSE NO. DPR-3

YANKEE ATOMIC ELECTRIC COMPANY

YANKEE NUCLEAR POWER STATION (YANKEE)

DOCKET NO. 50-29

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1.0 INTRODUCTION

By letter dated September 30, 1982 (as supplemented October 15, 1982 and November 10, 1982), Yankee Atomic Electric Company (the licensee) requested changes to the Technical Specifications for the Yankee Nuclear Power Station (Yankee). The purpose of the changes is to permit Yankee to operate with a reloaded core (Core XVI). The proposed changes also permit operation with changes to certain containment isolation valves, portions of the electric distribution system, and to the electric motor operators for certain valves.

2.0 BACKGROUND

The Yankee reactor core consists of 76 fuel assemblies, each having a 16 x 16 array of fuel rods. The Core XVI reloaded core will be loaded with a two region configuration of 40 fresh and 36 recycled assemblies fabricated by Exxon Nuclear Company. The licensee provided an extensive description of the replacement fuel in reports transmitted by letters dated July 4, 1975, and November 7, 1975, and we approved the use of the fuel in the Yankee reactor in License Amendment No. 21 dated December 4, 1975.

3.0 REACTOR PHYSICS

3.1 Introduction

The performance analysis of core reload XVI for the Yankee plant is described in the report YAEC-1325, September 1982, entitled "Yankee Nuclear Power Station, Core XVI, Performance Analysis," (Ref. 1) and its revision (Ref. 2). The description of the nuclear design and the analyses of the rod withdrawal transient, failure to borate prior to cooldown, the rod drop transient and the rod ejection accident have been reviewed.

3.2 Discussion and Evaluation

The physics characteristics of core XVI are similar to the corresponding characteristics of core XV. The main reason for it is the fact that the enrichment has not changed. The computational techniques used are the same as those used and approved previously. The radial power distribution is similar to the previous one with a maximum peak $F_{xy} = 1.862$ with the control rod group C fully inserted at 50[illegible] MWD/MTU. The disolved boron content for refueling at BOL is 1810 ppm versus 1746 ppm for core XV, which causes the moderator temperature coefficient to be less negative i.e., $-.92 \times 10^{-4}$ $\Delta p/^{o}F$ at hot full power. The effective delayed neutron fractions for both cores are essentially the same. The reactivity worth of the highest control rod is higher in core XVI than in XV due essentially to fuel arrangement. The analytical methods for cores XIV, XV, and XVI used the depletion calculations of PDQ/HARMONY with a few group cross sections derived using the LEOPARD program. SIMULATE was used to calculate the reactivity parameters such as moderator temperature coefficients, fuel termperature coefficient, boron worth and critical boron concentrations. Values measured at startup of cores XIV and XV were found to be in good agreement with SIMULATE predictions. The INCORE program was used for the analysis of flux measurements.

The slight differences in various parameters were consistent with the changes in the XVI cycle design. On this basis we find the nuclear design acceptable.

In the discussion of the transient and accident conditions the reference cycle bounds the values for core XVI except for the initial Minimum Departure from Nucleate Boiling Ratio (MDNBR) which is marginally less and the core inlet temperature which is 4^{o}F higher than the reference cycle XI. The impact of the small decrease in MDNBR will be discussed below separately for each event for which it is important. Operation with the higher core inlet temperature has been reviewed and approved in previous cycles and is still acceptable.

The values of the reactivity coefficients applied as input in the transient analysis are slightly different from the reference cycle and their impact will be discussed in each event for which it is important.

The effect of the reduced MDNBR limit on the control rod withdrawal is minimal. Analyses indicate that the MDNBR reamins above 2.0, (i.e., significantly greater than the safety limit of 1.3). The boron dilution transient analyses was referenced to cycle XV except for Mode 6 (re-fueling) for which additional shutdown margin was assumed to be derived from the actual boron concentration rather than that required for the 5% $\Delta\rho$ margin. In the case of failure to borate prior to cooldown the analysis indicated that core XVI is slightly more conservative than the reference core with respect to available shutdown margin. Likewise the important input parameters for the rod ejection accident are bounded by the most recent analysis of the event, i.e., that for core XIV. Based on the above discussion we find the accident analyses for core XVI to be acceptable for these events. We have reviewed the planned startup physics program and find that both the procedures and the acceptance criteria are similar to the NRC approved startup test programs and are therefore, acceptable.

3.3 Conclusions

We have reviewed the nuclear design transient characteristics and the startup physics program for the Yankee core XVI operation and find it acceptable.

4.0 REACTOR FUELS

4.1 Introduction

The Cycle XVI reload application involves a fuel design similar to that previously considered for Yankee. In addition to the changes in the fuel system design, the applicaiton also contains several minor revisions in the fuel design area of the plant safety analysis, as reported in (Ref. 1).

4.2 Fuel Systems Design

The Yankee Cycle XVI core will consist of 76 fuel assemblies with fuel rods arranged in 16 x 16 arrays. Each fuel rod is composed of a number of UO_2 fuel pellets in an eight-foot-long Zircaloy-4 tube. The outer row of fuel rods in each array are only partially filled to allow room for insertion of cruciform control rods between assemblies. The structural integrity of each assembly is maintained by six spacer grids and eight guide bars. The metal guide bars, or tie rods, replace fuel rods in the outside row of each assembly and provide structural support in the vertical direction. The assembly is not shrouded (Ref. 4), as was the case for several earlier fuel designs used in Yankee. The Yankee FSAR (Ref. 5) continues to show the older design. A discussion of the current design, including revised fuel design drawings, can be found in Reference 6.

4.3 Fuel Mechanical Design

The Cycle XVI core will consist of 40 fresh and 36 previously irradiated fuel assemblies. The spacer grids, grid straps, guide bars, instrumentation tubes and thimbles of all assemblies are now fabricated from Zircaloy-4 rather than 304 stainless steel. This design changes was approved (Ref. 7) for the previous cycle of operation. The change reduces parasistic absorption of neutrons and allows the use of slightly lower U^{235} content (3.5% vs. 4.0%) in the fuel.

4.4 Fuel Thermal Design

The fresh fuel in the Yankee Cycle XVI core is nearly identical to that previously irradiated in the reactor. The licensee's analysis of the fuel thermal performance is also the same as that used in the previous reload analyses including the consideration of power history effects and burnup-dependent fission gas release. No other changes in the methods used to analyze the fuel thermal design have been made. We therefore conclude that the fuel thermal design analysis for Yankee Cycle XVI is acceptable.

4.5 Transient and Accident Analysis

The analysis of transient and accident conditions for Yankee Cycle XVI generally follows methods and analyses previously approved by the NRC. Because a reanalysis of the postulated loss-of-coolant accident (LOCA) was required for Cycle XVI, we have examined the reload report to determine if fuel-related issues (e.g., cladding swelling and rupture, enhanced fission gas release) continue to be addressed in an appropriate manner. Our examination shows that the Yankee Cycle XVI core continues to satisfy our fuel-related requirements in a manner previously reviewed and approved by the staff.

4.6 Conclusions

We have reviewed the fuel system design and analysis for Yankee Cycle XVI operation and find it acceptable.

5.0 THERMAL-HYDRAULICS

5.1 Introduction

The Yankee Cycle XVI core contains 40 fresh fuel assemblies and 36 exposed assemblies, all fabricated by Exxon Nuclear. The core average burnup for the beginning of Cycle XVI is 5265 MWD/MTU compared to 6917 MWD/MTU for the cycle XV. Cycle XVI is the first core to contain a full core of assemblies with guide bars and spacer girds fabricated from Zircaloy-4 which was first introduced in Cycle XV. The use of Zircaloy-4 requires an increase in grid strap thickness by 3 mils.

5.2 Thermal-Hydraulic Evaluation

The thermal-hydraulic evaluation of the reload Cycle XVI has been performed utilizing basically the same methodology (i.e., the COBRA-IIIC thermal hydraulic code and the W-3 critical heat flux correlation) as in the previous cycles. Since a complete safety analysis was performed for Cycle XI, the Cycle XI serves as the reference thermal-hydraulic analysis as has been the case for Cycles XII through XV. The thermal-hydraulic analysis of the reload cycle was performed by adjusting code input to reflect the reload cycle power distributions and thermal-hydraulic characteristics.

A comparison of the thermal hydraulic design conditions for Cycle XVI and Cycle XI for 4-loop operation is provided in Table 1. Also listed in this table are the predicted values of the design parameters for the purpose of comparison against the design values. As in Cycle XV, the predicted hot channel factors are based on the beginning-of-life power distributions obtained when Rod Group C is 25 percent inserted even though rod restributions do not permit operation at full power in this mode. As shown in this table, the Cycle XVI has significant margin to DNB, coolant quality and fuel centerline melt limits. The design DNBR for Cycle XVI (3.20) is essentially the same as the DNBR for Cycle XI (3.24) at full power 4-loop operation. However, the Cycle XVI design DNBR is slightly larger than the DNBR for Cycle XV, i.e., 3.20 versus 3.13 at full power 4-loop operation. This slight difference occurs because of small difference in flow characteristics between Cycles XV and XVI. Cycle XVI has a full core of assemblies with Zircaloy grids and, therefore, the spacer gird straps for both fresh and exposed fuel have exactly the same thickness and the same hydraulic characteristics. In Cycle XV, only the fresh fuel assemblies have the thicker Zircaloy spacer grids which provides higher flow resistance. Therefore, the flow rate through the fresh fuel of Cycle XV was marginally less and resulted in lower DNRB.

The effect of fuel rod bow on DNBR was calculated by the method described in the Interim Safety Evaluation Report (Ref. 8) and applied a full gap closure (rod-to-rod contact) condition which resulted in a 34 percent DNBR reduction. The licensee in a letter dated February 9, 1977 (Ref. 9) described the available DNBR margins resulting from its design analysis. A generic DNBR margin of 13.2 percent was obtained from the following categories:

(a) Margin available resulting from the design nuclear enthalpy rise factor versus the calculated values.
(b) Margin available resulting from the design one pin peak power versus the calculated values.
(c) Margin available due to design conservatism applied to hot channel enthalpy rise.
(d) Margin available due to the design conservatism applied to one-pin peak power.

The DNBR credit of 13.2 percent had been reviewed and accepted by the staff for the previous cycles. Additional DNBR margin can also be obtained from the most limiting anticipated transient, which is the two of four pump loss of flow transient. Based on design conditions, the two of four pump loss of flow transient resulted in a miniumum DNBR in excess of 2.05. Therefore, the limiting transient has a DNBR margin of 36.6 percent compared to the DNBR limit of 1.3 for the W-3 correlation. The staff concludes that enough margin exists to offset the rod bow penalty.

With regard to transient analyses, the licensee has considered and compared the anticipated operations occurrences and the postulated accidents with the most recent appropriate analyses from the previous cycles, most of which were performed during Cycle XI. These analyses have been approved by the NRC previously. Table 7.1 of YAEC-1325 provides the initial operating conditions for most of the transients. Minor differences between the reload core and the reference cycle (Cycle XI) exist in basic plant parameters. These differences are as follows:

(a) Cycle XVI core inlet temperature is 4°F higher than the reference cycle. Plant operation at 515°F core inlet temperature was approved by the NRC during the Cycle XIII reload application. The reviews performed during the Cycles XIII through XV reload demonstrated the minor impact of the 4°F increase.
(b) Maximum design linear heat rate and hot channel factors are reduced from Cycle XI values. Cycle XVI values are identical to the values of Cycle XIII through XV. These values are more favorable than the reference Cycle XI values.
(c) Minimum DNBR at design condition for Cycle XVI is marginally less than the Cycle XI reference analysis value.

The staff has reviewed the initial conditions used in the transient analyses. The limiting transient for Yankee with respect to DNB is the loss of flow in two of four pumps which resulted in a minimum DNBR in excess of 2.05.

5.3 Conclusion

The staff has reviewed the thermal hydraulic design of Cycle XVI and finds that the difference in thermal hydraulic parameters between core XVI and the reference core XI is insignificant. The staff concludes that the thermal hydraulic design of core XVI meet the design criteria and is acceptable.

TABLE-1

Thermal Hydraulic Parameters for Yankee

Cycle XVI vs Cycle XI* During 4-Loop Operation

General Characteristics	Predicted	Design
Total Core Power, MWt	600 (600)	618 (618)
Main Coolant Pressure, Psig	2000 (2000)	1925 (1925)
Main Coolant Inlet Temperature, °F	515 (511)	519 (515)
Total Coolant Flow Rate, lb/hr	38.3×10^6 (38.3×10^6)	38.3×10^6 (38.3×10^6)
Nominal Channel Hydraulic Diameter, in	0.412 (0.399)	0.412 (0.399)
Average Mass Velocity, lb/hr-ft^2	2.29×10^6 (2.36×10^6)	2.29×10^6 (2.36×10^6)
Average Heat Transfer Area, ft^2	167 (171)	167 (171)

Hot Channel and Hot Spot Parameters

Maximum Centerline Pellet Temperature °F	2881 (3430)	3307 (3770)
Minimum W-3 DNB Ratio	4.43 (4.48)	3.20 (3.24)
Total Heat Flux Factor	2.41 (2.67)	2.76 (2.96)
Total Enthalpy Rise Factor	1.49 (1.76)	1.78 (1.81)

*Values in parentheses

6.0 TRANSIENT AND ACCIDENTS ANALYSES

6.1 Overview

The licensee reanalyzed the following accidents for Cycle XVI:

1. Boron Dilution Event
2. Isolated Loop Startup Incident
3. Loss of Load Incident
4. Loss of Feedwater Flow Incident
5. Loss of Coolant Flow Incident
6. Steam Line Rupture Incident
7. Steam Generator Tube Rupture Incident

For all the above transients either the reference cycle envelops the new conditions or the revised analyses presented in Cycle XVI submittal continue to show acceptable results when compared with the Standard Review Plan except for the Boron Dilution Event.

6.2 Boron Dilution

The analyses provided by the licensee for the boron dilution event present the time available to the operator prior to criticality from the time the dilution starts rather than from the time of an alarm.

For modes 4 and 5, the time available to the operator from the start of dilution is 25.4 minutes while the Standard Review Plan (SRP) requires 15 minutes from time of alarm till loss of margin. For mode 6 (refueling) the time indicated by the license analysis is 35.2 minutes while the SRP requires 30 minutes. Thus, unless the operator were to be alerted to the boron dilution immediately after the dilution is initiated, it is not clear how much time would be available to the operator to take the necessary action to terminate the event. Moreover, we do not know what signals or alarms are available or will be operable to alert the operator, and are unable to conclude that sufficient time exists for the operator to recognize the event and prevent recriticality. This issue is being considered generically by the staff for all PWRs. Because the consequences of such an event are small, the staff has deferred requiring licensees to meet the time limits specified in the SRP (Ref. 12), but will continue to study the subject. For this reason, we conclude that Yankee may continue operation during Cycle XVI, but note that additional analyses may be required as part of the resolution of the generic topic.

6.3 Main Steam Line Break

The Main Steam Line Break (MSLB) accident is under review by the staff separately, and our findings will be forwarded at the conclusion of the I&E Bulletion 80-04 review.

6.4 Loss of Coolant Accident

The licensee has used methods of analysis previously approved by the NRC in reevaluating the LOCA performance for Cycle XVI. These methods, when used within the limits specified for the analyses, have been shown to yield LOCA results that are in conformance with the limits specified in 10 CFR 50.46 and Appendix K, and they are therefore acceptable.

6.5 Conclusion

We conclude that operation of Yankee Cycle XVI is acceptable. The Boron Dilution Event will be addressed after the NRC finishes its generic review of the subject. The Main Steam Line Break will be addressed at a later date as part of I&E Bulletin 80-04 review.

7.0 SAFETY INSTRUMENTATION MODIFICATIONS

7.1 Discussion

As a part of the Environmental Qualification Program for Safety-Related Equipment, YAEC committed to replace Pressure Switch SI-PS-14. The switch is located inside the Yankee containment and is used to monitor main coolant pressure for the initiation of the Safety Inspection Actuation System - Train B (SIAS-B). This pressure switch function is being replaced by the output of a bistable in an existing analog pressure channel.

The licensee proposed changes to pages 3/4 3-12, 3/4 3-13, 3/4 3-14, and 3/4 3-15 of the Technical Specifications which would revise the nomenclature used to describe this function. For SIAS-B the "Low Main Coolant Pressure Sensor" would become the "RPS Low Main Coolant Loop 2 Pressure Channel," and on page 3/4 3-13, an additional paragraph has been added to note (3) to ensure consistent testing of the channels.

7.2 Evaluation

These changes do not affect any safety limits or reflect any change to a limiting safety setting and are essentially administrative in nature. The in-containment components of the new analog channel are qualified for their environment. The bistable portion is located in the main control board and as such, is in a mild environment. The new arrangement will be identical to that used for initiation of the Safety Injection Actuation System - Train A (SIAS-A). The analog channel supporting SIAS-B is independent of that used for SIAS-A. Each channel will be supplied by independent uninterruptible safety class power sources.

For these reasons, we conclude that these changes are acceptable.

8.0 CONTAINMENT ISOLATION VALVES

8.1 Introduction

As a result of modifications to various systems during the past refueling outage, boundary valves for containment isolation have been changed. The sections below describe the various changes for specific systems.

8.2 Charging System

8.2.1 Description

As a result of the Systematic Evaluation Program (SEP) seismic review, a new Hot Shutdown System has been proposed. During this refueling outage, pipe taps were installed to permit possible system installation during normal plant operation. Specifically the capped end of the alternate emergency feed header was removed and a normally closed 2-inch manual globe isolation valve, VD-V-1157, was installed and capped (see SK-82-19-2A).

8.2.2 Evaluation

As a result of this change, it was necessary to add containment valve VD-V-1157 to Table 3.6-1 of the Technical Specifications (TS). Valve SI-V-701 was also added to reflect the new boundary and valves VD-V-1093, 1094, 1095, and 1096 were deleted from the list. Because these valves are connected to the secondary side of the steam generators and are neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere, their use in this way meets the requirements of General Design Criterion (GDC) 57 of Appendix A to 10 CFR 50. We therefore conclude that this TS change is acceptable.

8.3 Atmospheric Steam Dump System

8.3.1 Description

The Atmospheric Steam Dump System (ASDS) was modified as part of the installation of the Post-Incident Cooling System (PICS). The PICS was designed and installed to ensure that safe shutdown and cooldown can be accomplished from within shielded areas in the highly unlikely event that a large source term is produced in the containment as a result of a major accident. Site radiation levels would prohibit significant operations from outside the shielded areas following an event. This system concept was developed as part of the Design Review of Plant Shielding performed under Item II.B.2 of NUREG-0737.

One of the functions required to maintain safe shutdown is the removal of decay heat. This will be accomplished by venting steam to atmosphere from the steam generators. The previous atmospheric steam dump utilizes manual valves requiring local operation at the Non-Return Valve (NRV) platform. This modification replaced the manual valves with motor-operated valves capable of operation from the Control Room. This will provide the capability to vent steam from the steam generators to the atmosphere for decay heat removal from within a shielded area.

The ASDS utilizes a single 2-inch line off of each main steam line. These lines tie into the main steam lines before the Non-Return Valves (NRV). Each line contains a Motor-Operator Valve (MOV) capable of remote operation from the Main Control Room. Each MOV is equipped with a manual handwheel for local operation. The four atmospheric steam dump lines are independent of one another; i.e., they are not connected by a common header. These lines discharge the steam directly to atmosphere.

With four 2-inch lines, the total steam release rate of the ASDS is adequate to remove decay heat immediately following a plant trip. The system also has adequate capacity to establish and maintain a 50°F/hr. cooldown rate.

8.3.2 Evaluation

This modification required the addition of four new containment isolation valves (MS-MOV-659, 670, 681, and 692) to Table 3.6-1 of the TS. These valves are connected to the secondary side of the steam generators and are neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere. We therefore conclude that their use in this fashion meets the requirements of GDC 57 and that this TS change is acceptable.

8.4 Appendix J Modifications

8.4.1 Description

On September 2, 1982 the staff issued its evaluation of Yankee's request for exemption from certain provisions of Appendix J to 10 CFR 50 for certain valves. Based on that evaluation, the licensee installed new trip valves, block valves, and test valves in the Component Cooling Water Supply, the Service Water Supply, and Vapor Container (VC) Heat Steam Supply lines. These valves will be testable in accordance with Appendix J requirements and will replace previously installed check valves which were not testable. Sketch 82-9-51 contains Figures A, B, and C, respectively, which illustrate the changes which were made.

Figure A is the Component Cooling Supply Line. CC-V-772 and CC-V-769 will be normally open. These block valves will be used for the testing of CC-TV-208. CC-V-770 and CC-V-771 are test valves that are normally closed.

Figure B is the VC Heat Steam Supply Line. HC-V-1200 is existing and will be closed, along with HC-V-1282 as block valves, during the testing of HC-RV-413. HC-V-1283 and HC-V-1284 are test valves and are normally closed.

Figure C is the Service Water Supply Line. Valves SW-V-812 and SW-V-813 are existing and will be closed with SW-V-1060 and used as block valves for testing of SW-TV-412. Valves SW-V-1058 and SW-V-1059 are test taps.

All block valves in these lines will be normally open.

Three new air-operated valves have been added to the Containment Isolation System (CIS). One of the valves, CC-TV-208, has been designated "essential;" the other two valves, SW-TV-412 and HC-TV-413, are "non-essential." "Essential" is defined as being required to be open during safety injection. All three valves have been incorporated into CIS in the same manner as the other CIS valves. Instrument air is provided from an air header in the upper PAB to each valve. Two solenoid valves have been placed in each line. One solenoid will energize on CIS "A" actuation; the other solenoid will energize on CIS "B" actuation. Energizing either solenoid valve vents air from the Air-Operated Valve (AOV) and the AOV closes. In addition to closing on CIS actuation, the "non-essential" valves will also close upon Safety Injection Actuation System (SIAS).

8.4.2 Evaluation

These modifications require the addition of valves TV-208, TV-412, and TV-413 to Table 3.6-1.

In addition, the following containment isolation valves are deleted due to the installation of the new valves:

CC-V-667	CC-V-660
CC-V-663	SW-V-820
CC-V-671	SW-V-821
CC-V-675	SW-V-822
CC-V-649	SW-V-823
CC-V-653	HC-V-1199

GDC 57 requires that "Each line that penetrates primary reactor containment and is neither part of the reactor coolant pressure boundary nor connected directly to the atmosphere shall have at least one containment isolation valve which shall be either automatic, or locked closed, or capable of remote manual operation." The Component Cooling Water lines, the VC Heat Steam Supply lines, and the Service Water lines are all part of systems which are closed inside the containment. The new valves will shut automatically on a containment isolation signal. The existing Component Cooling return line, the VC Heating Condensate return line, and the Service Water return line have existing automatic valves (TV-205, TV-409, and TV-408, respectively). We therefore conclude that the use of these valves in this fashion meets the requirements of GDC 57, and the changes to the TS are acceptable.

9.0 ELECTRICAL SYSTEM MODIFICATIONS

9.1 Introduction

A number of modifications have been made to the electrical distribution system during the Cycle XVI refueling outage at Yankee. In its letter dated September 30, 1982 the licensee requested approval of several Technical Specification changes which would be necessary as a result of those modifications. In Appendix A of that letter the licensee described each of the modifications being made. This section addresses the electrical portions of that appendix.

9.2 Station Batteries

In item 5 of the appendix the licensee indicated that batteries 1 and 2 have been replaced with larger batteries, and the existing motor-generator type battery chargers have been replaced with larger static type chargers. The new batteries are installed in the existing battery rooms on new two-step seismically qualified racks. The batteries are sized to supply their safety load for two hours and the new chargers are sized to supply the largest combined demand of the various stedy-state loads while recharging the batteries. We have reviewed the distribution system one-line diagrams, dc bus loads, and battery discharge load profile associated with this modification and find them acceptable.

9.3 Vital Busses

In item 6 of the appendix the licensee indicated that two new vital buses were installed to provide power to safety class instrumentation. The buses are powered by two new inverters having sufficient capacity to supply both the existing load and additional new instrumentation load. The inverters are powered from the new, larger batteries and chargers. Vital Bus 1 is powered from Battery 1 and Vital Bus 2 is powered from Battery 2. The additional dc bus loading due to the new inverters has been accounted for in the sizing of the new batteries and battery chargers. An alternate ac power source is provided to the vital buses from new Emergency MCCs 5 and 6. These MCCs take their power from new Emergency MCCs 3 and 4 which are in turn connected to existing Emergency Buses 1 and 2. The old instrumentation vital bus cabinet and emergency supply cabinet have been removed, and the remaining instrumentation power supplies were rearranged to supply the remaining instrumentation loads. We have reviewed the distribution system one-line diagrams associated with these modifications and find them acceptable.

9.4 Motor Operated Valve Dual Contactors

In item 7 of the appendix the licensee has identified a number of motor operated valves which will have an additional contactor wired in their starting circuits in order to meet the single failure criterion. These valves were originally disabled electrically by removing power leads from the motor starters. The additional starters were added to make the valves available during normal plant operation. The licensee has stated that the dual contactor arrangement is the same as that provided in other plant modifications. The staff approved the dual contactor design for those modifications in Amendment No. 52 and No. 69 to the Yankee operating license. The present modifications are, therefore, also acceptable.

9.5 Post-Incident Cooling System Modifications

In item 8 the licensee stated that as a result of the shielding review required by NUREG-0737, a Post-Incident Cooling System was installed which will permit operation of essential equipment from within shielded areas (Control Room and Switchgear Room). The modifications requiring technical specification changes include powering safety injection valves SI-MOV-48, 514, 515, 516, 517, and 518 from the new Emergency MCCs 3 and 4 and installation of an additional contactor for Safety Injection Tank Recirculation Valve CS-MOV-532 and Safety Injection Valve SI-MOV-4 to prevent spurious operation. The acceptability of the dual contactor arrangement has been discussed in the previous paragraph. The one-line diagrams for the new Emergency MCCs 3 and 4 have been reviewed and are acceptable.

9.6 Conclusions

We have reviewed the changes made to section 3/4.8.2 of the Yankee Technical Specifications and find that they result in no reduction of the existing Technical Specification requirements. Where vital power supplies have been added as a result of the above modifications the new Technical Specification addresses the addition with the same Limiting Conditions of Operation and Surveillance Requirements as required for existing vital power supplies. For the new batteries and battery chargers the operating parameters identified in the surveillance requirements have been changed to agree with the new system requirements. The new Technical Specification also allow the battery performance discharge test to be performed in lieu of the battery service test once per 60 month interval rather than requiring both tests every 60 months. This is consistent with new Standard Technical Specification requirements. We therefore conclude that the changes made to the Yankee Technical Specifications as a result of these modifications are acceptable.

10.0 SUMMARY OF FINDINGS

From our review of the material submitted by the licensee on the Core XVI reload, we find:

A. The mechanical design of the fuel, the nuclear and thermal-hydraulic analysis, and the analyses of accidents and transients are acceptable, except that certain aspects of the Boron Dilution Event and the Main Steam line Break accident will be evaluated separately by the staff.

B. The modifications to the electric motor operators, to the electrical distribution system, and to the containment isolation valves are acceptable.

C. The proposed Technical Specifications, which implement the electrical and other changes, and which modify the reactor core operational limits, are acceptable.

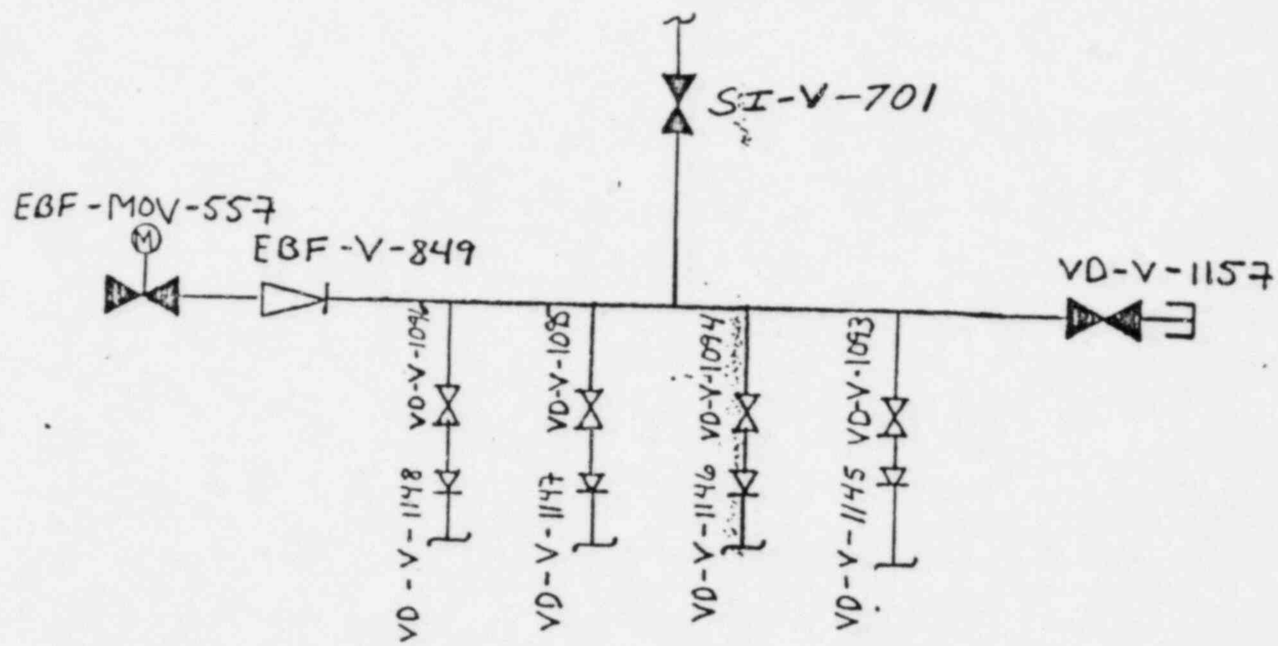
11.0 ENVIRONMENTAL CONSIDERATION

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and pursuant to 10 CFR 51.5(d)(4) that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of the amendment.

12.0 CONCLUSION

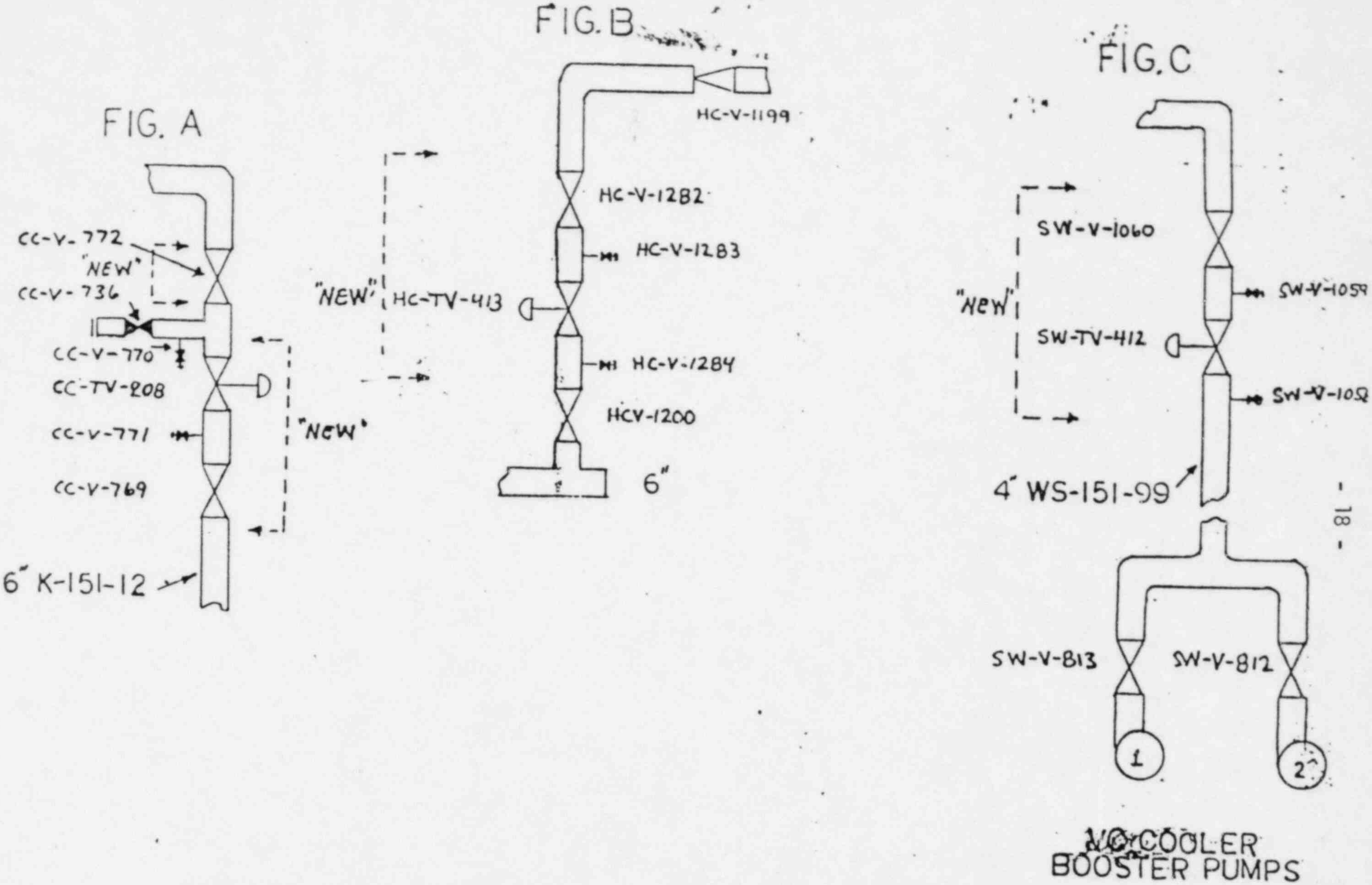
We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security of to the health and safety of the public.

FIGURE 13.1



YANKEE ATOMIC ELECTRIC COMPANY NUCLEAR SERVICES DIVISION	DRAWN BY T. Schmidt CHECKED BY	TITLE. Flow Diagram - Alternate Emergency Feed Header
DWG. NO. SK - 82-19-2A	APPROVED BY	

FIGURE 13.2



YANKEE ATOMIC ELECTRIC COMPANY
NUCLEAR SERVICES DIVISION

DRAWN BY R.D.L. 7-7-82

CHECKED BY

APPROVED BY

TITLE. VALVE ARRANGEMENT

DWG. NO. SK 82-9-S1

14.0 REFERENCES

1. J. A. Handschuh, et al, "Yankee Nuclear Power Station Core XVI Performance Analysis," YAEC-1325, dated September 1982.
2. J. G. Robinson, et al, "Revision to Cycle XVI, Core Performance Analysis," YAEC-1325, dated November 10, 1982.
3. L. H. Heider (YAEC) letter to ONRR, (NRC) on "Core XVI Refueling" dated October 15, 1982.
4. J. L. French (YAEC) letter to ONRR, (NRC) dated July 15, 1975.
5. Yankee Nuclear Power Station FSAR, YAEC Report dated January 3, 1974, U. S. NRC Docket Number 50-29.
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8. D. F. Ross and D. G. Eisenhut (NRC) memorandum for D. B. Vassallo and K. R. Goller (NRC) "Interim SER on the Effects of Fuel Rod Bowing on Thermal Margin Calculations for Light Water Reactors" dated December 8, 1976.
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