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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION RELATED TO AMENDMENT NO. 16 TO FACILITY OPERATING LICENSE DPR-70

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
PHILADELPHIA ELECTRIC COMPANY
DELMARVA POWER AND LIGHT COMPANY
ATLANTIC CITY ELECTRIC COMPANY
SALEM NUCLEAR GENERATING STATION, UNIT NO. 1
DOCKET NO. 50-272

Introduction

By letters dated June 29, September 25, 1978, and February 16 and April 24, 1979, Public Service Electric and Gas Company (the licensee) proposed changes to the Technical Specifications appended to Operating License No. DPR-70 for the Salem Nuclear Generating Station Unit No. 1. The proposed changes would (1) clarify the moderator temperature coefficient specification, (2) revise the requirements for the performance of Type C containment isolation valve leakage tests, (3) decrease the required range of the condenser outlet temperature detectors, (4) conform the containment structural integrity surveillance to 10 CFR 50 Appendix J requirements, and (5)include various administrative changes.

Discussion

Moderator Temperature Coefficient

By letter dated June 29, 1978, the licensee proposed to revise section 3.1.1.4 of the Technical Specification such that the limiting condition for operation (LCO) for moderator temperature coefficient (MTC) would be based on the all rods out configuration for the end of cycle life (EOL) which is equivalent to the MTC for all rods in configuration for EOL now specified as the LCO. This in effect would require that the MTC be equal to or less than -3.8×10^{-4} delta k/k/°F for all rods out at EOL and rated thermal power rather than equal to or less than -5.0×10^{-4} delta k/k/°F for all rods in at EOL and rated thermal power. This would be a non-substantive change which would bring this LCO into conformance with the current versions of the Westinghouse Standard Technical Specifications (STS). Consistent with the current STS, we also suggested certain changes to the licensee's proposed ACTION statements that would be observed should the MTC LCO not be met. These changes would require that the appropriate control rod withdrawal limits be established and maintained to restore the MTC to within the LCO limit within 24 hours. The licensee has agreed to these changes.

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Condenser Outlet Temperature Detectors

Also in its letter of June 29, 1978, the licensee proposed to decrease the required range of the condenser outlet temperature detectors from that specified in the Environmental (Appendix B) Technical Specifications (0 -150°F, with an instrument accuracy of +0.5°) to 32°F -150°F. This change is justified on the basis that there is no need for calibrated indication below the freezing point of water, 32°F. This is an administrative change having no safety or environmental impact. Since it will have no effect on actual operations, no further evaluation of this change is required.

Containment Type C Leakage Tests

In its letter of September 25, 1979, the licensee proposed to eliminate the requirement to perform Type C leakage tests on 8 valves in the main steam system and 8 valves in the feedwater system. The basis for this proposal is that these valves are not intended to isolate containment in the event of a steam-line break accident, but are intended to isolate main steam and feedwater in the event of such an accident. The licensee has proposed that the valves identified be tested at least once per 18 months only to ensure that they actuate to the isolate position on a main steam isolation test signal.

The licensee has also proposed to eliminate the requirement to perform Type C leakage tests on both 1CV68 and 1CV69 and would instead perform such tests on either one of these valves. The basis for this request is that these valves are redundant isolation barriers outside containment on a 3" CVCS charging line whereas Appendix J of 10 CFR 50 only requires a single isolation barrier on either side of containment.

Containment Structural Integrity

By letter dated February 16, 1979, the licensee proposed to change the surveillance requirements for the visual inspection of the containment structure to be consistent with the Westinghouse STS. The proposed change would combine the surveillance requirements for concrete surfaces and the containment liner plate and would change the requirement that the visual inspection be performed during the Type A containment leakage rate tests while the containment is at its maximum test pressure. Appendix J requires that such an inspection be performed as a prerequisite to the Type A tests.

Miscellaneous Administrative Changes

In Amendment No. 5 issued cn May 31, 1977, for the Salem Unit No. 1 license, we changed section 4.10.2 to allow a thermal power limit increase for Physics Tests from 85% to 100% of rated thermal power specifically during the Power Coefficient and Load Swing Tests performed as part of the initial Startup Test Program. Since that program has been completed, this one-time provision is no longer appropriate and can therefore be removed. Since this decrease in allowable power during physics tests can have no adverse impact on plant safety, no further evaluation of this change is required.

We recommended that the description of the fuel assemblies as contained in section 5.3 of Appendix A to the license be revised to specify the maximum total weight in grams of uranium (1766) per fuel rod rather than the nominal weight (1743). This change would more accurately reflect staff requirements and would be consistent with the Westinghouse STS.

Evaluation

Moderator Temperature Coefficient (MTC)

The proposed change clarifies an existing ambiguity in the LCO for MTC by stating the position of all the control rods. Since the currently approved STS for Westinghouse plants state the MTC limit for all rods out position, that position has been proposed. The existing LCO is based on the all rods in position; therefore a simple conversion was performed to obtain the equivalent MTC for the desired all rods out position. Since the licensee's proposed change and those changes we suggested with regard to the required ACTION statements would simply bring the Salem Unit No. 1 Technical Specifications for MTC into agreement with the Westinghouse STS for more recently approved facilities, they are considered to be administrative in nature, to have no adverse impact on safety, and are therefore acceptable.

Containment Type C Leakage Tests

Main Steam and Feedwater System Valves

The eight main steam valves (11-14 MS7 and 11-14 MS18) and the eight feedwater valves (11-14 BF 19 and 11-14 BF40), for which the licensee proposes to delete Type C leakage test requirements, are designed to isolate the main steam system and feedwater system, respectively, in

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response to a steam-line break accident. These valves are not identified in the Salem Final Safety Analysis Report (FSAR) as required to isolate containment. Therefore, there is no requirement, pursuant to Appendix J of 10 CFR 50, to conduct Type C leakage tests on the valves identified. The licensee has therefore proposed to delete that requirement and to add a surveillance requirement that would verify only that each main steam isolation valve actuates to its isolation position on a main steam isolation test signal. This test would be performed at least once per 18 months which is consistent with the isolation test for the feedwater isolation valves that already exists in the Salem Technical Specifications.

In our independent assessment of potential releases of radioactivity following either a main steam line break accident or steam generator tube rupture accident, we concluded, prior to licensing, that the consequences of these accidents can be controlled by limiting the permissible reactor coolant system and secondary coolant system radioactivity concentrations so that potential offsite doses are well within 10 CFR Part 100 guidelines. Technical Specification Limiting Conditions for Operation (LCO's) therefore exist specifying limits for reactor coolant system specific activity, and allowable reactor coolant system leakage rates.

To ensure that the assumptions used in our LOCA analyses remain valid, we requested that the licensee reconfirm that no other leakage path (not subject to Type C leakage tests) would exist from the containment environment through the main steam system. By letter dated April 24, 1979, the licensee indicated that the only potential paths for leakage from the containment atmosphere to the main steam system are through four one-inch hydrotest vent connectors (one located at the high point of the main steam line from each of the four steam generators) and possibly through the steam generator tubes themselves. As indicated in the licensee's letter, the hydrotest vent connections are isolated during normal operations by two one-inch manual globe valves with a cap tack-welded on the nipple of the outermost valve. With regard to potential leaks through the steam generator tubes, the licensee indicated that station procedures require that following LOCA, the steam generator water levels will be maintained such that the tube bundles would be covered and potential leaks prevented.

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In view of the above, we concluded that the assumptions considered in determining the potential consequences of the design basis accidents discussed above would remain valid and that the deletion of Type C leakage tests for main steam valves 11-14 MS7 and 11-17 MS18 and feedwater valves 11-14 BF-19 and 11-14 BF40 is therefore acceptable.

Chemical and Volume Control Systems (CVCS) Valves

General Design Criterion (GDC) 55 of Appendix A to 10 CFR 50, Reactor Coolant Pressure Boundary Penetrating Containment, requires (among other things) that each line that is part of the reactor coolant pressure boundary and that penetrates primary reactor containment shall be provided with containment isolation valves outside containment consisting of one locked closed or one automatic isolation valve. Appendix J to 10 CFR 50 requires that containment isolation valves be periodically subject to Type C leakage tests to demonstrate the ability of the valves to provide adequate containment isolation in the event of an accident. Since GDC 55 does not require two isolation valves outside containment (it requires one inside and one outside), valves 1CV68 and 1CV69 listed in Table 3.6-1 of the Salem Technical Specification should not both be required to be subject to a periodic Type C leakage test. The licensee has therefore proposed to add a footnote to Table 3.6-1 that would require that either 1CV68 or 1CV69 be subject to Type C leakage test, but not both valves. However, since section 3.6.3.1 requires that all valves in Table 3.6-1 be OPERABLE, and since section 3.6.1.2 (Containment Leakage) identifies the maximum combined leakage rate for the valves subject to Type C tests, as identified in Table 3.6-1, we proposed that the footnote should simply require that either valve 1CV68 or 1CV69 must be OPERABLE. This would necessitate that both the Type C leakage test and the closing capability be satisfied for whichever of the two valves the licensee chooses to designate as the containment boundary for that CVCS line that penetrates containment. The licensee agreed to this change to its proposal. In the course of our review of this request, the licensee identified six check valves that had been inadvertently excluded from Table 3.6-1. These valves are in the CVCS component

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cooling water and pressurizer relief tank systems and constitute the containment isolation function inside containment as required by GDC 55. The licensee has agreed that these valves should be added to Table 3.6-1 of the enclosed Technical Specifications thus including these valves with the others requiring periodic (once each 24 months) Type C leakage tests. The licensee has indicated that the six valves will be leak tested and demonstrated OPERABLE during the current refueling outage. We find that the changes discussed above are in compliance with GDC 54, 55 and 57 as appropriate, and conform to Appendix J of 10 CFR 50. We, therefore, conclude they are acceptable.

Containment Structural Integrity

The proposed revision to the surveillance requirements of Section 3.6.1.6, Containment Structural Integrity, clarifies the inspections to be performed, the items to be checked during the inspections and when the inspections are to be performed. We find that the proposal is consistent with Appendix J to 10 CFR 50 and with the staff-approved Westinghouse STS. We therefore find this proposal to be acceptable.

Miscellaneous Changes

The revised description of the fuel assemblies as contained in Section 5.3, more explicitly defines the uranium content of the Salem fuel assembly rods. The revised wording defines a maximum fuel content instead of a nominal value and is consistent with the staff-approved Westinghouse STS. The licensee has agreed to this change. Although Section 5.3 is intended to provide a basic description of the design features and not necessarily to identify parameters used in the Safety Analysis, a maximum fuel content has more meaning than a nominal value and is therefore more desirable. The change is considered to be administrative in nature and is therefore acceptable.

Environmental Consideration

We have determined that this 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 this amendment.

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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 or to the health and safety of the public.

Date: June 1, 1979

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RELATED TO AMENDMENT NO. 16 TO FACILITY LICENSE NO. DPR-70

SALEM NUCLEAR GENERATING STATION, UNIT NO. 1

DOCKET NO. 50-272

INTRODUCTION

By letters dated March 1 and April 19, 1979, Public Service Electric and Gas Company (the licensee) proposed changes to the Technical Specifications appended to Facility Operating License DPR-70 for the Salem Nuclear Generating Station, Unit No. 1. The proposed amendment would permit removal of the part-length control rods. This has been done on other Westinghouse reactors.

DISCUSSION AND EVALUATION

The Technical Specifications, as now written, require that these part-length rod cluster control assemblies (PLRCCAs) be withdrawn and excluded from the core at all times during reactor operation. The PLRCCAs are not needed, used or assumed to be available in any safety analysis of the facility. The proposed removal, therefore, will not cause any change in required reactivity characteristics or safety margins at full power, low power or shutdown. To the contrary, removal will eliminate the potential for part-length rod insertion into the core during operation. Such an event could cause an abnormal flux distribution or reactor shutdown.

In order to preserve the current dynamic operating characteristics of the reactor (i.e., pressure drops, coolant flow rates, etc.) which could be affected if just removal of the PLRCCAs were to be performed, the licensee proposes to install thimble plug assemblies in the spaces previously occupied by PLRCCAs. The thimble plug assembly consists of a flat base plate with short rods suspended from the bottom surface and a spring pack assembly. The twenty short rods, called thimble plugs, project into the upper ends of the guide thimbles to reduce the bypass flow area. Fuel assemblies without control rods, burnable poison rods, or source rods use identical devices. Similar short rods are also used on the source assemblies and fuel assembly guide thimbles. As installed in the core, the thimble plug assemblies interface with both the upper core plate and with the fuel assembly

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top nozzles by resting on the adapter plate. The spring pack is compressed by the upper core plate when the upper internals assembly is lowered into place. Each thimble plug is permanently attached to the base plate by a nut which is locked to the threaded end of the plug by a pin welded to the nut.

All components in the thimble plug assembly, except for the spring, are constructed from type 304 stainless steel. The springs are wound from Inconel X-750 for corrosion resistance and high strength.

The thimble plugs will effectively limit bypass flow through the rod cluster control guide thimbles in the fuel assemblies from which the PLRCCAs have been removed, just as they currently limit bypass flow in those assemblies which do not contain control rods, source rods, or burnable poison rods.

Based on the considerations that (1) the PLRCCAs are not needed for reactor operation, (2) that removal of these assemblies will remove the chance for an abnormal flux distribution or reactor shutdown and (3) that insertion of the thimble plug assemblies will preserve the current dynamic operating characteristics of the reactor, we conclude that this change is acceptable.

ENVIRONMENTAL CONSIDERATION

We have determined that this 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 this 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 this amendment.

CONCLUSION:

We have concluded, based on the considerations discussed above, that: (1) because the amendment involves neither a significant increase in the probability or consequences of accidents previously considered nor 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 the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: June 1, 1979