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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 81 TO FACILITY OPERATING LICENSE NO. DPR-50

METROPOLITAN EDISON COMPANY
JERSEY CENTRAL POWER AND LIGHT COMPANY
PENNSYLVANIA ELECTRIC COMPANY
GPU NUCLEAR CORPORATION

THREE MILE ISLAND NUCLEAR STATION, UNIT NO. 1

DOCKET NO. 50-289

Introduction

By letter dated November 10, 1981 (L1L 314, TSCR No. 103), GPU Nuclear Corporation (the licensee) requested a change to the Three Mile Island Nuclear Station, Unit No. 1 (TMI-1) Technical Specifications (TSs). This change would add additional TSs to permit performance of a Low Power Natural Circulation Test following the restart of TMI-1. The licensee committed to perform this test in response to NUREG-0694, Item I.G.1, which was imposed by the NRC staff on TMI-1 by letter dated November 25, 1980, and reaffirmed by the Atomic Safety and Licensing Board (ASLB) in its December 14, 1981, Partial Initial Decision (PID) in the TMI-1 restart proceeding (paragraph 1132).

Discussion and Evaluation

As part of the low power testing program for the restart of TMI-1, the licensee proposes to operate the reactor outside the range of the current plant TSs and requests that the TSs be modified to encompass the conditions of the proposed tests. The tests are designed to demonstrate the ability of the primary and secondary system to remove energy from the core at decay heat power levels and to provide operator training. Since the decay heat generation rate at TMI-1 is very low because of the long shutdown period, the reactor will be brought critical and operated at a power level of approximately 3% of design (2535 MWT) to simulate decay heat. This modification to the TSs will be cancelled at the completion of the test program.

Six tests are proposed which are briefly summarized below:
1. Determination of indicated reactor power conversion factor. This test will be performed with the reactor coolant pumps in operation. The average reactor system temperature will be reduced to 526°F. The current TSs require that the reactor be subcritical if the reactor coolant temperature is below 525°F. Although the licensee does not believe that this limit will be violated, a reduction of the limit to 520°F is requested to reduce the potential for a TS violation.

2. Emergency feedwater actuation test. This test is designed to demonstrate that following a loss of main feedwater event, the emergency feedwater pumps will start automatically and emergency feedwater flow will be properly indicated within the control room. The reactor coolant pumps will remain in operation. No TS changes are required for this test.

3. Verification of two-hour air supply capability of the bottled backup air supply. In this test the ability of the backup air supply to control air operated emergency feedwater system valves will be verified. The reactor coolant pumps will remain in operation. No TS changes are required for this test.

4. Establishment of Natural Circulation Flow. This test will verify that the reactor can be controlled manually in natural circulation. For this test, the reactor trip logic will be modified so that the reactor will not trip on loss of forced coolant flow. TS changes are required to permit bypassing this trip function.

5. Verification of the adequacy of the pressurizer heaters on the emergency bus. This test will verify the adequacy of the pressurizer heaters to maintain natural circulation at hot standby with only emergency power available. The heaters will then be de-energized and the effect on natural circulation will be monitored as the reactor system decreases to 1800 psig. At that pressure, the test will be terminated. The reactor trip logic will be modified so that the reactor may be operated at low power without forced reactor coolant flow and so that the reactor will not trip on low pressure. TS changes are required to permit bypassing the automatic trip functions associated with loss of forced flow or low pressure.

6. Determination of the effect of decreased once-through steam generator levels on natural circulation flow. In this test, the secondary side water level will be slowly decreased to 90 inches on the startup range and the effect on natural circulation recorded. If natural circulation is lost, the secondary level will be increased so that natural circulation is reestablished. The reactor trip logic will be modified so that the reactor can be operated at low power without forced reactor coolant flow. TS changes are required to permit bypassing this trip function.

The low power tests proposed at TMI-1 are designed to be performed at power levels simulating decay heat. Since the amount of decay heat at TMI-1 is very low, the reactor will be operated at approximately 3% of full power. Administrative controls will require that the reactor be manually tripped at 5%, of full power.

For the natural circulation tests, the licensee will lower the power level trip setpoint to 7% of full power, and additional administrative controls will be placed on reactor system pressure and temperature. A 7% power level compares to the decay heat level immediately after reactor trip from full power operation. Similarily, a 5% power level would be present after 15 seconds and a power level of 3% would be present after 200 seconds following a reactor trip from full power. Analyses in

Chapter 14 of the TMI-1 Final Safety Analysis Report (FSAR) indicate that forced reactor coolant flow is not required at these power levels to prevent fuel damage. The analyses further demonstrate that with the steam generator water level at the design value of 50% on the operating range, natural circulation will be adequate to transfer the heat generated by the core to the steam generators.

In Test Number 6, the steam generator water level will gradually be lowered to determine the effect on natural circulation. During this test, natural circulation may be lost. Loss of natural circulation will be indicated to the operator by increasing primary system temperature and pressure and by reduced steam and feedwater flowrate in the steam generators. The operator will increase steam generator level if natural circulation is lost and will trip the reactor if the subcooled margin is lost. Tripping the reactor will reduce the reactor heat generation rate to a very low level since little fission product inventory will be built up in the core. Even in the event that the reactor system became saturated and boiling occurred in the core, fuel damage would not occur at the test power levels. This is demonstrated by analyses of small break Loss of Coolant Accidents which show that following a reactor trip from full power, the core will be adequately cooled as long as it remains covered with water or a two-phase mixture[1]. We have evaluated potential design basis transients and accidents and believe that with the additional requirement discussed below, the consequences of any design basis event at the low power levels of the proposed tests will be less than the consequences of transients and accidents analyzed in Chapter 14 of the FSAR. For the natural circulation tests (Tests 4, 5 and 6), reduction of the overpower setpoint to 7% will provide additional protection for transients and accidents which produce an increase in power level, such as inadvertent control rod withdrawal.

For Test Number 1, the reactor coolant temperature will be lowered below the values evaluated in the FSAR. The licensee has not evaluated the effect of increased system stiffness produced by reduced temperature for a potential control rod withdrawal event. We, therefore, require that the reactor overpower setpoint be set at 7% for this test as well as for Tests 4, 5 and 6. The licensee has agreed to implement this additional requirement and TS 3.20.1.1 has been revised from the licensee's proposed TS to reflect this.

Since the licensee has agreed to establish the reactor overpower trip setpoint at 7% for Test Number 1, the reduced trip setpoint will be available for all tests which require modification of the current TSs. This automatic protection, as well as constant operator surveillance combined with administrative controls on power level, temperature, pressure

[1] Letter from J. Taylor, B&W, to S. Varga, NRC, attaching additional Emergency Core Cooling System small break analyses, July 18, 1978.

and subcooling margin set forth in the test procedures will provide adequate protection for the reactor core and reactor system pressure boundary during the performance of these low power tests. On this basis, we find the proposed change to the TSs acceptable.

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 this amendment.

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 an accident previously evaluated, does not create the possibility of an accident of a type different from any evaluated previously, and does not involve a significant reduction in a margin of safety, 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.

Dated: FEB 28 1983

The following NRC personnel have contributed to this Safety Evaluation: W. Jensen, R. Jacobs.