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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 78 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

I. Introduction

By letters dated May 18, 1981 (L1L 125, TSCR No. 103), June 10, 1981 (L1L 167), August 4, 1981 (L1L 183), November 13, 1981 (L1L 329, TSCR No. 103, Rev. 1), and May 20, 1982 (TSCR No. 103, Rev. 2) Metropolitan Edison Company[1/] proposed changes to the Technical Specifications (TSs) appended to Facility Operating License No. DPR-50 for the Three Mile Island Nuclear Station, Unit No. 1 (TMI-1). The changes involve the incorporation of certain of the TMI-2 Lessons Learned Category "A" requirements and other short-term requirements identified in the Commission Order of August 9, 1979.

II. Background

By letter dated September 13, 1979, we issued to all operating nuclear power plants requirements established as a result of our review of the TMI-2 accident. Certain of these requirements, designated Lessons Learned Category "A" requirements, were to have been completed by the licensee prior to any operation subsequent to January 1, 1980. In the case of GPU Nuclear, the Lessons Learned Category "A" requirements and other short-term requirements addressed by the licensee's application were imposed on the licensee as a result of the August 9, 1979 Commission Order (hereinafter referred to as "the Order"). Our evaluation of the licensee's compliance with these requirements is provided in NUREG-0680, "TMI-1 Restart Evaluation" and Supplements Nos. 1, 2 and 3. The licensee's compliance with and the staff's evaluation of these requirements was a subject of the TMI-1 restart proceeding.

In order to provide reasonable assurance that operating reactor facilities are maintained within the limits determined acceptable following the implementation of the TMI-1 Lessons Learned Category "A" items, we requested that all licensees amend their licenses to incorporate additional Limiting Conditions of Operation and Surveillance Requirements, as appropriate. For GPU Nuclear, this request was made part of the Commission Order of August 9, 1979. GPU Nuclear's application is in response to this Order. Each of the issues identified in the licensee's application is addressed below.

1/ GPU Nuclear is now the licensed operator for TMI-1.

III. Evaluation

A. Anticipatory Reactor Trips (ARTs)

One of the short-term Order requirements was IE Bulletin 79-05B. This bulletin required that the licensee provide a safety grade automatic ART using appropriate signals. The ART proposed by the licensee would trip the reactor upon loss of main feedwater or upon a turbine trip. By tripping the reactor on the anticipation of a pressure transient, (1) the probability of an overpressure trip is reduced, and (2) the challenge rate to the pressurizer code safety valves is reduced. The challenge rate to the pressurizer power operated relief valve (PORV), a piece of equipment which is not credited in the safety analysis, is likewise reduced. The design of the reactor trips on loss of feedwater or turbine trip incorporates a 2-out-of-4 logic, is fully testable, and meets the single failure criterion of IEEE-279.

A description of the licensee's design of this reactor trip system has been submitted as part of the licensee's "Report In Response to NRC Staff's Recommended Requirements for Restart of TMI-1" (Restart Report) (Section 2.1.1.1).

Our evaluation of the licensee's submittal was reported in NUREG-0680 (p. C2-12 to C2-14). We found that the licensee's design of the ART was acceptable and agree that this change will reduce the probability of an overpressure reactor trip, reduce the challenge rate to the pressurizer code safety valves, and reduce the challenge to the PORV, thereby enhancing the plant's safety margin. The licensee's proposed change to the TSs provides similar limiting conditions for operation and surveillance requirements as those required for other reactor protection system components. For the above reasons, we find the licensee's proposed TS changes for the ART acceptable.

B. Operability of PORV and Block Valve, Position Indicators for PORV and Safety Valves and Setpoints

Category "A" requirement 2.1.3a of NUREG-0578 required the licensee to install position indication to be able to unambiguously determine the position of the PORV and safety valves. As described in Section 2.1.1.2 of the Restart Report, the safety and relief valve position indication system consists of the following:

(1) delta-pressure taps and monitors, at discharge piping elbows downstream of the safety and relief valves, and

(2) acoustic monitor (accelerometer) mounted on the pressurizer PORV.

The above sensors are in addition to the tailpipe thermocouples and reactor coolant drain tank instrumentation which is presently installed. Our evaluation of this system is reported in NUREG-0680 (p. C8-11 through C8-13) and NUREG-0680, Supplement 3 (p. 26, Item 2.1.3a). We found the licensee's proposed system acceptable in these evaluations.

These indications are a diagnostic aid for the plant operator and provide no automatic action. The licensee has provided TSs which require the primary position indication for these valves to be operable to allow continued operation of the plant. Surveillance requirements include a channel check every shift and refueling interval calibration for the delta-pressure monitors and monthly channel test and refueling interval calibration for the acoustic monitor. We find the proposed TSs for this instrumentation acceptable.

The licensee has also proposed TS additions to help ensure the operability of the PORV and block valve. Whenever the PORV is determined to be inoperable, the associated block valve must be shut and electrically isolated to allow continued operation. Likewise if the PORV block valve is inoperable, the PORV shall be closed and electrically isolated. Continued operation is permitted with the PORV and/or block valve shut since no credit is taken for these valves in safety analyses (except for plant low pressure protection which was addressed in Amendment No. 56 dated July 28, 1980). Surveillance requirements include cycling the block valve on a quarterly basis and a monthly channel test and refueling interval calibration of the PORV setpoints. These surveillance requirements are consistent with Standard Technical Specifications (STS) for these valves and are thus acceptable.

C. Emergency Power Supply Requirements (Pressurizer Heaters)

Category "A" requirement 2.1.1 of NUREG-0578 required the licensee to ensure that sufficient pressurizer heaters could be connected to emergency power supplies to support natural circulation in the event of loss of offsite power. In the Restart Report, Section 2.1.1.3, the licensee proposed a design which allows manual realignment of two pressurizer heater banks (126 KW) to onsite emergency power supplies in the event of loss of offsite power. Our evaluation and acceptance of the licensee's design is reported in NUREG-0680 (p. C8-6 to C8-8). The licensee has proposed TSs to require operability of a minimum of 107 KW of heaters (the conservatively calculated pressurizer heat loss) and operability of the emergency power supplies for the two heater banks in order to allow continued operation of the plant. For surveillance requirements, the licensee has proposed a refueling interval demonstration of transfer of pressurizer heaters to their emergency power supplies which is consistent with STS and considered acceptable to provide reasonable assurance that the minimum number of pressurizer heaters will be available to support natural circulation.

D. Containment Isolation

Category "A" requirement 2.1.4 of NUREG-0578 required the licensee to provide diverse signals for containment isolation of nonessential systems under postulated accident conditions. In Section 2.1.1.5 of the Restart Report, the licensee provided details of a redesigned containment isolation system with the following new features:

(1) Partial containment isolation on reactor trip
(2) Partial containment isolation on 30 psig building pressure
(3) Specific line isolation on high radiation
(4) Specific line isolation on rupture detection
(5) Specific line isolation on 1600 psig Reactor Coolant System (RCS) pressure

As reported in NUREG-0680 (p. C8-21 to C8-26) and NUREG-0680, Supplement 3 (p. 30 to 33), the revised containment isolation system provides diverse isolation provisions for nonessential systems, and containment isolation will not be automatically lost after the containment isolation signal is reset. Therefore, we have found the licensee's revised containment isolation system acceptable.

The licensee has proposed TSs to require operability of two channels of each containment isolation signal (except high radiation) whenever plant conditions require containment isolation to be in effect. The high radiation signal on the purge line is considered a backup to the other diverse signals of reactor trip and 4 psig reactor building pressure. Hence, redundant requirements for this signal are unnecessary. For surveillance requirements, the licensee has proposed a channel check every shift, and channel test monthly when containment integrity is required in addition to a refueling interval calibration on all signals. These TS requirements are similar to those imposed on other safety-related instruments and systems and are acceptable.

E. Instrumentation to Detect Inadequate Core Cooling

Category "A" requirement 2.1.3b of NUREG-0578 required the licensee to install a primary coolant saturation meter to provide on-line indication of coolant saturation condition. In the Restart Report, Section 2.1.1.6, the licensee provided a description of the proposed meter. The saturation meter will continuously display in the control room, the margin between primary plant temperature (T_{HOT}) and the saturation temperature (Tsat) for existing RCS pressure. An alarm will be initiated if the margin falls below a pre-set value. Redundancy will be provided by computing and displaying Tsat margin independently for each reactor coolant loop. Our evaluation and acceptance of the proposed meter is reported in NUREG-0680 (p. C8-16 to C8-19).

The licensee has proposed TS additions to require operability of at least one saturation meter to allow continued plant operation. Since the saturation meters perform no automatic safety function and since backup methods (such as steam tables) are available in the event the saturation meters are inoperable, we find this acceptable. For surveillance, the licensee has proposed surveillance requirements consistent with those required for safety-related instrumentation which we find acceptable.

F. Emergency Feedwater System Requirements

Items 1 and 8 of the August 9, 1979 Commission Order addressed modifications and procedural changes which would enhance Emergency Feedwater (EFW) system timeliness and reliability. As described in the Restart Report, Section 2.1.1.7, modifications to the EFW system include (1) upgrading power supplies to the motor driven EFW pumps; (2) providing for automatic initiation of EFW pumps on loss of both feedwater pumps or loss of all reactor coolant pumps; and (3) providing steam generator level indication, EFW flow indication and manual EFW control independent of the Integrated Control System (ICS). Procedure modifications and training are also being developed to ensure EFW is available and properly applied when required. Our evaluation of these modifications and procedural items as well as additional requirements identified as significant in EFW system reliability studies, is provided in NUREG-0680 and NUREG-0680, Supplement 3. We found the licensee's proposed modifications and procedural changes acceptable in our evaluation.

To help ensure the availability of the EFW system, the licensee has proposed TS additions to require that all EFW pumps and associated flow paths be operable whenever the RCS is above 250°F. If pumps and/or flow paths are inoperable, specified time periods are prescribed to bring the plant to cold shutdown depending on how many pumps and flow paths are inoperable. Additional operability requirements are placed on the EFW pump automatic initiation features and the EFW flow instrumentation. The operability requirements described above are consistent with those requirements placed on other engineered safety features (such as Emergency Core Cooling System (ECCS)) and are therefore acceptable.

With regard to surveillance requirements, the licensee has proposed TS additions to require periodic EFW pump testing every 31 days, valve lineup verification every 31 days and periodic flow tests following cold shutdown outages of greater than 30 days and at least every refueling interval. Automatic initiation circuitry is tested quarterly and calibrated each refueling interval. Operability of the manual control valve station is also verified quarterly. EFW flow instruments are tested monthly and calibrated each refueling interval. The above surveillance requirements exceed those required by STS and are therefore acceptable.

G. RCS Pressure Setpoints

IE Bulletin 79-05B required that the licensee propose modifications to reduce the likelihood of automatic actuation of the PORV during anticipated transients. In response to this bulletin, the licensee has raised the PORV setpoint from 2255 to 2450 psig RCS pressure and lowered the high RCS pressure reactor trip setpoint from 2390 psig to 2300 psig. The licensee referenced a supporting generic Babcock & Wilcox Company (B&W) analysis which was reviewed by the NRC staff. The results of the NRC staff's evaluation of this B&W analysis were issued as NUREG-0565, "Generic Evaluation of Small Break Loss of Coolant Accident Behavior in Babcock & Wilcox Designed 177-FA Operating Plants", dated January 1980. Additional TMI-1 specific evaluation is provided in NUREG-0680 (p. C1-12 to C1-16). The licensee has proposed TS changes which incorporate the revised PORV actuation and high RCS pressure trip setpoints which, based on the above analysis and evaluation, we find acceptable.

In addition, the reactor protection system low RCS pressure trip setpoint per existing TSs is 1800 psig. The B&W generic ECCS analysis, "ECCS Analysis of B&W's 177-FA Lowered Loop NSS", BAW-10103, Rev. 2, references a value of 1900 psig for the low RCS pressure trip setpoint based on trip on variable pressure/temperatures at full power. For operation at less than full power, the proposed TS change increases the minimum allowable low pressure trip from 1800 to 1900 psig.

The purpose of the low RCS pressure trip is to maintain thermal margins for the fuel by preventing the minimum Departure from Nucleate Boiling (DNB) ratio from decreasing below the safety limit of 1.3. Increasing the low RCS trip setpoints to 1900 psig has the effect of increasing the margin to DNB following a trip on low pressure (because the reactor trips earlier in a transient). Thus the final minimum DNB ratio would be higher (more conservative) than if the reactor tripped at 1800 psig. On this basis, we find the proposed TS change setting the minimum low RCS pressure trip setpoint at 1900 psig, 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

These TS changes were required as a result of the Commission's August 9, 1979 Order and the proceeding initiated thereby to decide if restart should be authorized. The TS changes proposed by the licensee are an adequate response to the Commission's directive to incorporate additional limiting conditions for operation (LCOs) and surveillance requirements (SR). For this reason and for other reasons specified in our evaluation above, we conclude that there is reasonable assurance that the activities authorized by the above TS changes can be conducted without endangering the health and safety of the public and 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: October 20, 1982

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