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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

LOW TEMPERATURE OVERPRESSURE MITIGATING SYSTEM

FOR

THREE MILE ISLAND NUCLEAR STATION, UNIT NO. 1

SUPPORTING AMENDMENT NO. 56 TO

FACILITY OPERATING LICENSE NO. DPR-50

METROPOLITAN EDISON COMPANY

DOCKET NO. 50-289

1.0 Introduction

By letter dated October 15, 1976 (Reference 1) Metropolitan Edison Company (Met-Ed) submitted to the NRC a plant-specific analysis in support of the reactor vessel overpressure mitigating system (OMS) for Three Mile Island Unit 1 (TMI-1) Nuclear Power Station. The analysis was supplemented by letter dated March 22, 1977 (Reference 2) and other documentation submitted by Met-Ed (References 3-6). Met-Ed has installed the equipment and incorporated the procedures described in this report. Hence, this report summarizes past efforts by the licensee, vendor, and NRC staff.

NRC Staff review of all information submitted by Met-Ed in support of the proposed overpressure mitigating system is complete and has found that the system provides adequate protection from overpressure transients. A detailed safety evaluation follows.

2.0 Background

Over the last few years, incidents identified as pressure transients have occurred in pressurized water reactors. This term "pressure transients," as used in this report, refers to events during which the temperature pressure limits of the reactor vessel, as shown in the facility Technical Specifications, are exceeded. All of these incidents occurred at relatively low temperature (less than 200 degrees F) where the reactor vessel material toughness (resistance to brittle failure) is reduced.

The "Technical Report on Reactor Vessel Pressure Transients" in NUREG 0138 (Reference 7) summarizes the technical considerations relevant to this matter, discusses the safety concerns and existing safety margins of operating reactors, and describes the regulatory actions taken to resolve this issue by reducing the likelihood of future pressure transient events at operating reactors. A brief discussion is presented here.

2.1 Vessel Characteristics

Reactor vessels are constructed of high quality steel made to rigid specifications, and fabricated and inspected in accordance with the time-proven rules of the ASME Boiler and Pressure Vessel Code. Steels used are particularly tough at reactor operating conditions. However, since reactor vessel steels are less tough and could possibly fail in a brittle manner if subjected to high pressures at low temperatures, power reactors have always operated with restrictions on the pressure allowed during startup and shutdown operations.

At operating temperatures, the pressure allowed by Appendix G limits is in excess of the setpoint of currently installed pressurizer code safety valves. However, most operating PWRs did not have pressure relief devices to prevent pressure transients during cold conditions from exceeding the Appendix G limit.

2.2 Regulatory Actions

By letter dated August 11, 1976 (Reference 8), the NRC requested that Met-Ed begin efforts to design and install plant systems to mitigate the consequences of pressure transients at low temperatures. It was also requested that operating procedures be examined and administrative changes be made to guard against initiating overpressure events. It was felt by the staff that proper administrative controls were required to assure safe operation for the period of time prior to installation of the proposed overpressure mitigating hardware.

Met-Ed responded (Reference 1) with information describing measures to prevent these transients along with some discussion of proposed hardware. The proposed hardware change was to install a low pressure actuation setpoint on the existing pressurizer power operated relief valves.

Additional NRC staff concerns were expressed in letters to Met-Ed, dated December 9, 1976 and November 11, 1977 (References 9,10), respectively. Met-Ed responded to these concerns in References 2 through 6. The correspondence focused on system design criteria discussed below.

2.3. Design Criteria

Through this series of meetings and correspondence with PWR vendors and licensees, the staff developed a set of criteria for an acceptable overpressure mitigating system. The basic criterion is that the mitigating system will prevent reactor vessel pressures in excess of these allowed by Appendix G. Specific criteria for system performance are:

1) Operator Action: No credit can be taken for operator action for ten minutes after the operator is aware of a transient.

2) Single Failure: The system must be designed to relieve the pressure transient given a single failure in addition to the failure that initiated the pressure transient.

3) Testability: The system must be testable on a periodic basis consistent with the system's employment.

4) Seismic and IEEE 279 Criteria: Ideally, the system should meet seismic Category I and IEEE 279 criteria. The basic objective is that the system should not be vulnerable to a common failure that would both initiate a pressure transient and disable the overpressure mitigating system. Such events as loss of instrument air and loss of offsite power must be considered.

We also instructed the licensee to provide an alarm which monitors the position of the pressurizer relief valve isolation valves, along with the low setpoint enabling switch, to assure that the overpressure mitigating system is properly aligned for shutdown conditions.

Licensees were informed that their proposed mitigating systems were to meet these criteria for the most adverse of hypothesized scenarios, that is, the largest mass or heat addition which could occur at the specific plant. While administrative procedures were to be employed to reduce the probability of an initiating event, administrative procedures were not to be employed in lieu of hardware modifications. These hardware modifications were to provide sufficient relief capacity to mitigate the most adverse scenario.

It was recognized that these criteria were of a general nature and that exceptions would be required as individual reviews progressed. (See Section 3.1 Evaluation)

2.4 Design Basis Events

The incidents that have occurred to date have been the result of operator errors or equipment failures. Two varieties of pressure transients can be identified: a mass input type from charging pumps, safety injection pumps, safety injection accumulators; and a heat addition type which causes thermal expansion from sources such as steam generators or decay heat.

No overpressure event at low temperature has occurred at a B&W supplied NSSS. The most common cause of overpressure transients to date has been isolation of the letdown path. The staff has identified the most limiting mass input transient to be inadvertent injection by the largest safety injection pump. The most limiting thermal expansion transient is the start of a reactor coolant pump with a large temperature difference between the water in the reactor vessel and the water in the steam generator.

Met-Ed has provided an evaluation of:

a. Erroneous actuation of the High Pressure Injection (HPI) system.

b. Erroneous opening of the core flood tank discharge valve.

c. Erroneous addition of nitrogen to the pressurizer.

d. Makeup control valve (makeup to the RCS) and or its bypass valve fails full open.

e. All pressurizer heaters erroneously energized.

f. Temporary loss of the Decay Heat Removal System's capability to remove decay heat from the RCS.

g. Thermal expansion of RCS after starting an RC pump due to stored thermal energy in the steam generator.

3.0 System Description

The overpressure mitigating system (OMS) consists of active and passive subsystems. The active subsystem is simply the modification of the actuation circuitry of the existing electrical pilot operated relief valve (PORV) to provide dual setpoints, a normal operation setpoint of 2450 psig and a low pressure setpoint of 485 psig. The low pressure setpoint is employed when the reactor coolant system is below 275°F. This system is manually enabled. An alarm will function should the operator fail to enable the system. An alarm has also been installed to monitor the position of the pressurizer relief block valve, RC-V2. The passive subsystem consists of the introduction of a nitrogen blanket at the top of the pressurizer. The reactor is operated during heatup and cooldown with a steam or nitrogen bubble. The bubble functions as a mechanical damper. This subsystem is part of the original B&W design.

3.1 System Evaluation

The TMI-1 OMS is both redundant and functionally diverse. The plant, by virtue of a gas (nitrogen or steam) blanket in the pressurizer and the relatively small size, and hence heat capacity, of the once through steam generators, is not susceptible to heat addition transients. The plant is never operated in a water solid condition.

In contrast, the OMS of a Westinghouse or Combustion Engineering supplied NSSS consists of two relief valves with independent low setpoint actuation circuitry. The two trains are identical, i.e., not diverse. (It is noted the diversity although desirable was never an NRC staff design criteria.) These systems are susceptible to heat addition transients. These systems are operated in a water solid condition.

Met-Ed has submitted analyses of the design bases events shown in Section 2.4 (Reference 2). We accept these analyses. These analyses show that, in the event, of a postulated mass addition, actuation of the relief valve will limit RCS pressures to the relief valve setpoint and hence below Appendix G limits. Should the relief valve fail closed, or actuation circuitry fail, the system pressure would continue to increase. With the exception of postulated high pressure safety injection, the nitrogen bubble in the pressurizer will provide at least ten minutes, and in some cases substantially longer time, for operator action. The analyses also show that in the event that decay heat removal was lost, more than 15 minutes would pass before the relief valve setpoint would be reached. Postulated reactor coolant pump starts with steam generator secondary water temperature greater than primary water temperature will not result in RCS pressure increases to the relief valve setpoint value. Hence, TMI-1 is not considered susceptible to overpressure transients due to inadvertent heat addition.

System pressure overshoot, that is, increase of primary coolant pressure after pressure reaches the low setpoint value, does not occur on B&W supplied NSSS due to the rapid action of the electrically operated PORV and the relatively slow rates of pressure increase due to the nitrogen blanket in the pressurizer.

The TMI-1 OMS is tolerant of seismic events. Met-Ed has performed analyses for the pilot assembly connection pipe assuming seismic motion of 3.0g horizontal and 3.0g vertical. The actual valve meets Class 1 requirements. Testing with simulated seismic loadings has not been performed. This was not a requirement at the time the plant was designed and constructed. Even if it is assumed that the valve, connection pipe, or actuation circuitry failed due to a seismic event, the nitrogen blanket in the pressurizer would provide protection for postulated low temperature overpressure events.

The system is testable and is to be tested prior to use. The PORV is to be tested each shutdown.

The system does not strictly meet IEEE 279 criteria. The basic objective of this criterion, prevention of common mode failure, is met by virtue of the subsystem diversity.

For all postulated heat addition transients and for all mass additions other than inadvertent high pressure safety injection, the TMI-1 OMS meets single failure and operator action criteria.

In the event that the largest possible mass addition were to occur, one high pressure injection train, actuation of the relief valve would terminate the transient. Should this valve fail the RCS pressure would exceed system pressure in five to eight minutes (depending on the initial system conditions). Hence, for this postulated event, the system does not meet single failure/operator action criteria. For lesser mass addition rates, in the event that the relief valve failed, the pressurizer bubble would act as a pressure damper providing more than ten minutes for operator action.

In contrast, the OMS of a Westinghouse or Combustion Engineering supplied NSSS will (with specific plant exceptions), assuming that one of the ~~two~~ relief valves or associated circuitry were to fail, terminate this transient.

Administrative controls to mitigate high pressure injection must be found acceptable or additional hardware installed. Both options were considered and are discussed below.

A makeup/charging pump is run to provide RCS main coolant pump seal water. Actuation of a high pressure injection train consists of opening a high pressure injection (HPI) motor operated valve, MOV, permitting flow from the makeup/charging pump to the RCS. Circuit breakers for the closed HPI MOVs are "racked out" and "tagged" during plant cooldown. With the motor operator "racked out" flow through the valve would represent a passive failure and need not be considered. One must insure that these valves are closed when HPI is not needed without decreasing the probability that they can be opened when HPI is needed.

Options considered by Met-Ed include: modification of the decay heat removal system, modification of the makeup and purification system, addition of a second pressure relief valve on the pressurizer. These options were estimated (by the licensee) to cost $200,000 to $400,000. These options introduce additional safety concerns.

Relief capacity addition to the decay heat removal (DHR) system is only of value with respect to low temperature overpressurization when the DHR is aligned. This system is automatically blocked at a RCS pressure of 400 psig. Modification of the system would requi·e modification of the DHR autoclosure interlocks. Spurious failure of these modified interlocks would increase the probability of primary breaks outside of containment. Installation of relief and block valves downstream of the HPI valves (that is, modification of the makeup system) would increase the probability that HPI, if required, would be impaired. Addition of a second power operated relief valve on the pressurizer would increase the probability of a small break loss of coolant accident.

Hence, although these hardware modifications would comply with the letter of our guidelines, they are not considered advisable. Administrative controls supplemented by the single pressure relief train, and pressure and level indication and alarms, is considered a suitable and prudent alternative.

Credit for administrative controls is consistent with past staff actions. We have permitted a manually enabled system, credit for blocking safety injection actuation signal, credit for successfully blocking one of two high pressure safety injection trains, and credit for blocking accumulator injection. On Combustion Engineering and Westinghouse supplied NSSS we have assumed administrative control of the primary to secondary differential temperature for heat addition analyses. For B&W supplied NSSS, we have assumed that the nitrogen bubble will be established (a manual procedure) and that the initial pressurizer level will be controlled.

3.2 Electrical Controls

In addition to the above design features we recommended modifications to the monitoring system. These recommended modifications will alert the operator in the control room and allow time for taking corrective actions in preventing over pressurization of RCS at low temperature. These modifications which have been accepted by the licensee and will be in place prior to restart consist of:

1. In order to ensure HPI valves are "racked out" an alarm will be installed to alert the operator in the control room. The alarm will sound if the HPI valves are not "racked out" when the reactor coolant temperature is below 275°F. This alarm signal is bypassed during normal depressurization at a pressure of 1750 psig.

2. A pressure alarm will be installed in the control room alerting the operator when the reactor coolant pressure exceeds 485 psig and the temperature is less than 275°F.

3. Low pressure and temperature recorders will be provided to permanently record all low temperature/pressure transients.

4.0 Administrative Controls

To supplement the hardware modifications and to limit the magnitude of postulated pressure transients to within the bounds of the analysis provided by the licensee, a defense in depth approach is adopted using procedural and administrative controls. Specific conditions required to assure that the plant is operated within the bounds of the analysis are described below.

4.1 Procedures

A number of provisions for prevention of pressure transients are incorporated in the plant operating procedures.

(1) The OMS is to be manually enabled when the reactor coolant system temperature is less than 275°F. The low pressure setpoint is 485 psig which is a requirement incorporated in the plant Technical Specifications. An alarm will sound if the operator fails to enable the system. An alarm will also be actuated if the operator closes the PORV isolation valve and the RCS temperature is below 275°F.

(2) The plant is to be operated with a steam or nitrogen blanket in the pressurizer during plant cooldowns and heatups. The initial pressurizer water level is to be less than or equal to the high level alarm at system pressures above 100 psig and less than the high high level alarm for pressures less than or equal to 100 psig.

(3) The makeup tank water level is to be less than the high level alarm.

Extensive use of alarms insures that the operator is aware of vital plant conditions outside the bounds of those assumed in the safety analysis. The operator must take corrective action to clear these alarms. Overpressurization of the vessel might occur only if an initiating event was coincident with ignoring these alarms.

(4) Core Flood Tank discharge valves are to be closed and circuit breakers for the motor operators "racked out" during plant cooldown before the RCS pressure is decreased to 600 psig. This is normal procedure.

(5) High pressure injection motor operated valves are "racked out" during plant cooldown prior to startup of the Decay Heat Removal System. Startup of this system normally occurs at an RCS temperature of 250°F. If the reactor head is installed and the reactor coolant temperature is less than or equal to 275°F, that is, when the low temperature overpressure mitigating system is required to be operable, the high pressure injection pump breakers are to be racked out (these pumps supply water to the high pressure injection motor operated valves) unless the high pressure motor operator valves are closed and the pressurizer level is less than or equal to 220 inches. (This level is 40 inches below the high level alarm). This latter requirement is to be incorporated in the plant Technical Specifications.

(6) Testing which requires flow through the high pressure injection motor operated valves is only to be performed with the reactor vessel head off or the reactor coolant temperature greater than 320°F. This requirement is also to be incorporated in the plant Technical Specifications.

We find that the procedural and administrative controls described above are acceptable.

4.2 Technical Specifications

To ensure operation of the low temperature overpressure mitigating system, and decrease the probability that an initiating event which will challenge the system occurs, Met-Ed has proposed (Reference 6) to incorporate essential procedures discussed above (Section 4.1) in the plant Technical Specifications. These proposed changes to the plant Technical Specifications are acceptable.

5.0 Conclusion

The administrative controls and hardware changes proposed by the Metropolitan Edison Company provide protection for Three Mile Island Unit 1 from pressure transients at low temperatures by reducing the probability of initiation of a transient and by limiting the pressure of such a transient to below the limits set by Appendix G. We find that the overpressure mitigating system is acceptable as a long term solution to the problem of overpressure transients.

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 environ- 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 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.

Dated: July 28, 1980

References:

1. R. C. Arnold, "Evaluation of Potential Vessel Overpressurization," October 15, 1976, Met-Ed letter GQL 1448.

2. R. C. Arnold, "Evaluation of Potential Vessel Overpressurization," March 22, 1977, Met-Ed letter GQL 0332.

3. R. C. Arnold, letter to R. W. Reid, NRC, April 6, 1977, Met-Ed letter GQL 0464.

4. J. G. Herbein, "Overpressure Protection Systems," January 13, 1978, Met-Ed letter GQL 0049.

5. J. G. Herbein, "RCS Overpressurization," March 13, 1978, Met-Ed letter GQL 0426.

6. J. G. Herbein, "Technical Specification Change Request No. 74," March 13, 1978, Met-Ed letter GQL 0395.

7. "Staff Discussion of Fifteen Technical Issues listed in Attachment G, November 3, 1976 Memorandum from Director NRR to NRR Staff," NUREG-0138, November 1976.

8. R. W. Reid, NRC letter to Met-Ed, Re: Reactor vessel overpressurization in pressurized water reactor facilities, August 11, 1976.

9. R. W. Reid, NRC letter to Met-Ed, Re: Additional information to evaluate overpressure mitigating system, December 9, 1976.

10. R. W. Reid, NRC letter to Met-Ed, Re: Preliminary review of overpressure mitigating system, November 11, 1977.