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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 28 TO FACILITY OPERATING LICENSE NO. NPF-3

THE TOLEDO EDISON COMPANY

AND

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

DAVIS-BESSE NUCLEAR POWER STATION, UNIT NO. 1

DOCKET NO. 50-346

Introduction

By letter dated October 1, 1976 (Reference 1), the NRC notified the Toledo Edison Company (TECo or the licensee) that operating reactors' history had shown an unexpectedly large number of reported instances of reactor vessel overpressure events in Pressurized Water Reactors (PWR's) wherein Technical Specification (TS) limits implementing 10 CFR Part 50 Appendix G limitations had been exceeded. The majority of these cases had occurred during cold shutdown when the primary systems were in water-solid conditions. These overpressure events had been initiated by a variety of causes; but in essentially all of the cases reported, a single personnel error, equipment malfunction, or procedural deficiency was sufficient to cause the event.

In Reference 1, the NRC requested that TECo 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 considered by the NRC 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.

TECo responded (Reference 2) with information describing measures that would be taken at Davis-Besse, Unit No. 1 (DB-1) to prevent these transients. Additional NRC staff concerns were discussed at a meeting with TECo on February 17, 1977, at which time TECo attempted to justify operation with its proposed overpressure mitigation system. Subsequent TECo submittals (References 3 and 4) documented responses to the NRC's concerns and provided additional information about procedural controls, hardware, and TSs. The TECo proposal consisted of using the decay heat removal (DHR) system relief valve, which was sized to accommodate the most limiting overpressure transient. To assure that this relief path was always present during shutdown operations, TECo proposed removing power from the DHR isolation valves (DH 11 and DH 12) so that inadvertent closure could not take place. The proposed removal of power from these valves posed a problem area with the NRC due to a staff position that the DHR isolation valves should always receive a signal to close if system pressure reaches a high value.

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During the review of TECo's application for an operating license, the NRC staff was also concerned about the potential for inadvertent closure of the DHR isolation valves during operation of the system. Since DB-1 utilizes a single DHR suction line from the reactor coolant system (RCS) serving the two otherwise independent DHR trains, a single failure causing one of the valves to shut would remove suction from both trains, thus potentially damaging the system pumps.

To assure resolution of the NRC staff's concerns about inadvertent closure of the DHR isolation valves and about the potential for overpressure events, the DB-1 operating license was issued on April 22, 1977 (Reference 5) with a number of conditions including the following:

2.C.(3)(d) Prior to startup following the first (1st) regularly scheduled refueling outage, Toledo Edison Company shall install a long-term means of protection against reactor coolant system overpressurization.

2.C.(3)(j) Until such time as final resolution is obtained regarding the potential for and consequences of an inadvertent closure of a decay heat removal system valve during shutdown operations, Toledo Edison Company shall maintain power on decay heat removal isolation valves DH 11 and DH 12 and shall operate one decay heat removal train at a time.

2.C.(3)(o) Prior to entering Mode 5 (Cold Shutdown), Toledo Edison Company shall make a modification which ensures that the decay heat removal relief valve would actuate prior to automatic closure of the isolation valves. This change will allow the relief valve to be available for mitigating the consequences of an overpressure event.

Amendment No. 2 to the license (Reference 6) deleted condition 2.C(3)(o) after TECo modified the automatic closure setpoint of the DHR isolation valves to a value which was 93 psig above the DHR relief valve setpoint. Amendment No. 3 to the license (Reference 7) revised condition 2.C(3)(j) to read:

2.C.(3)(j) Until such time as final resolution is obtained regarding the potential for and consequences of an inadvertent closure of a decay heat removal system valve during shutdown operations, Toledo Edison Company shall maintain power on decay heat removal isolation valves DH 11 and DH 12 and shall operate one decay heat removal train at a time.

This license condition shall not preclude performance of specific surveillance or preoperational test requirements related to this equipment and associated instrumentation as provided in the Technical Specifications.

> For those periods of time during which only one decay heat removal train is available for operation or during the time that the standby decay heat removal train is being brought on line, an operator shall be stationed in the control room so as to immediately secure the reactor heat removal pump(s) should loss of flow occur due to the inadvertent closure of DH 11 and DH 12.

This safety evaluation addresses the resolution of conditions 2.C.(3)(d) and 2.C.(3)(j).

Background

The NRC staff report "Reactor Vessel Pressure Transient Protection for Pressurized Water Reactors", NUREG-0224 (Reference 8) 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.

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, prior to 1977 most operating PWRs did not have pressure relief devices to prevent pressure transients during cold conditions from exceeding the Appendix G limit.

Through a series of meetings and correspondence with PWR vendors and licensees, we developed a set of criteria for an acceptable overpressure mitigating sytem. The basic criterion is that the mitigating system will prevent reactor vessel pressures in excess of those 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.

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.

The incidents that had occurred at the time Reference 8 was prepared were 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.

Only one overpressure event at low temperature (during hydrostatic test) had occurred at a Babcock and Wilcox (B&W) nuclear supplied steam system (NSSS). The most common cause of overpressure transients was isolation of the letdown path. We 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.

TECo has provided evaluations for inadvertent actuation of the high pressure injection (HPI) system, thermal expansion of the RCS after starting a reactor coolant pump (RCP) due to stored thermal energy in the steam generator, and failure of the makeup control valve in the full open position. The potential for dumping of the core flood tanks was not analyzed by TECo since it was not considered a credible event.

System Description and Operation

The DB-1 system utilizes a relief valve (PSV 4849) in the DHR system suction line to provide overpressure protection when the RCS temperature is less than 280°F. The system will be placed into operation during plant cooldown when RCS temperature is between 340°F and 280°F and pressure is less than the relief setpoint of 320 psig. This will be accomplished by opening the DHR isolation valves DH 11 and DH 12 and removing power from their motor operators Position control of these valves, as well as the capability to remove power from the motor operators is available in the control room. Also, an alarm is provided in the control room anytime DH 11 or DH 12 is open and power has not been removed from the motor operators. The instrumentation and control for the alarm is safety grade. Valve position indicator is available from the control room whether or not power is provided to the motor operators.

Plant cooldown and depressurization will continue with DH 11 and DH 12 open and incapable of inadvertent closure. When pressure is decreased to less than 30 psig, the pressurizer steam bubble is replaced with nitrogen. Except for hydrostatic testing, the system is never allowed to go water-solid.

During plant heatup and repressurization, a steam bubble is drawn in the pressurizer and the nitrogen is vented when RCS pressure is greater than 50 psig. When RCS temperature is greater than 280°F, power is restored to the motor operators of DH 11 and DH 12 and the valves are closed. To ensure that both valves are closed before system repressurization, an interlock is provided that will trip the pressurizer heaters when pressure reaches 438 psig and either of the valves is not closed. If neither valve is closed, pressurization cannot occur above the PSV 4849 setpoint of 320 psig. Also, if power is provided to the motor operator of DH 11 and DH 12, an automatic closure signal will be sent to the valves if pressure reaches or exceeds 438 psig.

Valves DH 11 and DH 12, as well as their control systems, and relief valve PSV 4849 are seismically qualified. As an operator aid, a computer alarm is provided in the control room anytime RCS pressure approaches the TS limit closer than 200 psig.

The removal of power from DH 11 and DH 12 during shutdown allows credit for a protection device not vulnerable to a single active component failure (PSV 4849) and which could accommodate an inadvertent overpressure transient. The safety valve has been sized for the pressure surge resulting from actuation of two HPI pumps. Also, the licensee has stated that this safety valve will be tested to assure operability and proper set pressure during each refueling outage. We have reviewed the licensee's evaluation of pressure transients and based on these analyses conclude that an inadvertent actuation of the HPI pump would cause the worst credible pressure transient conditions while the reactor is starting up or shutting down and, therefore, conclude that the licensee's sizing requirements are conservative.

The pressure transients have been evaluated for the cases of having power removed and restored to the DHR isolation valves. Water-solid conditions were not assumed because a nitrogen blanket or a steam bubble is to be maintained in the pressurizer during cold conditions. The licensee has shown that the RCS pressure for reactor coolant temperatures greater than 280°F will not exceed the Appendix G limit (effective for the first five full-power reactor years) following an overpressure event with DH 11 and DH 12 in a closed position. For the case of having the DHR isolation valves in an open position, and power removed, the integrity of the DHR system following an overpressure event will be maintained by PSV 4849.

With DH 11 and DH 12 open and power removed, no operator action is required to ensure overpressure protection. Valves DH 11, DH 12, and PSV 4849 are seismically qualified, and the controls and alarms for DH 11 and DH 12 as well as the pressurizer interlock meet IEEE 279 criteria.

Based on the above, we find the licensee's proposed design to ensure against low temperature overpressure events at DB-1 to be acceptable. In addition, we conclude that sufficient administrative controls exist to minimize the likelihood of an overpressure event. Installation of the system and implementation of the administrative controls will be subject to verification by the Office of Inspection and Enforcement. On this basis, license condition 2.C.(3)(d) may be deleted. We concur in the licensee's conclusion that, with the relief capacity and setpoint of PSV 4849, the pressure of the DHR system can never exceed design pressure with DH 11 and DH 12 open and power removed from their motor operators. With power restored to the valves, the interface criteria of having two valves in series to separate the high pressure from the low pressure boundary will be met. Therefore, removing power from the opened valves DH 11 and DH 12 is not contrary to the NRC staff position that these valves should receive an automatic closure signal whenever the system pressure reaches a high value. License condition 2.C.(3)(j) may therefore be deleted from the license.

The proposed overpressure protection system has been analyzed by TECo to be acceptable for only the first five effective full-power years. After this time, the pressure-temperature limit curves shift enough to require additional pressure relief protection prior to aligning the RCS to the DHR system. We will require the licensee to submit proposed modifications to the overpressure protection system at the time that revisions to the TS pressure-temperature curves are submitted to the NRC for approval.

Technical Specifications

In Reference 4, TECo proposed TS changes summarized as follows:

a) Addition of operability and surveillance requirements for the pressurizer heater interlock.

b) Addition of operability and surveillance requirements for PSV 4849.

c) A change in the setpoint of the automatic closure signal for valves DH 11 and DH 12 from a value of >413 psig to a value of <438 psig. This value of <438 psig is also established as the setpoint for the pressurizer heater interlock.

The previous setpoint for the automatic closure signal of DH 11 and DH 12 of > 413 psig was based on the NRC staff requirements that power remain on the valves during DHR operation and that the valves should receive a signal to close when system pressure reaches a high value. The TS changes will require that DH 11 and DH 12 be open with their power removed whenever RCS temperature is less than 280°F, assuring a relief path to PSV 4849. Therefore, a lower limit on the automatic closure setpoint is no longer appropriate. Rather, an upper limit needs to be established to assure that DH 11 and DH 12 cannot be inadvertently opened when RCS pressure exceeds the design rating of the DHR system. The value of <438 psig selected by the licensee is based on DHR design pressure, allowances by the ASME Code, instrument string drift, and the difference in the static head between the pressure sensing point and the midpoint of DH 11 and DH 12. This value is conservatively chosen and is acceptable.

In our review of the overpressure protection system, we considered the proposed TSs necessary, but not sufficient. Additional requirements are necessary to minimize the potential for overpressure transients. These include a requirement for a pressurizer bubble to exist in conjunction with the operability of PSV 4849, a requirement to vent the RCS if PSV 4849 becomes inoperable, and a special reporting requirement if the overpressure protection system is ever challenged. By letter dated July 22, 1980, (Reference 9), TECo has committed to propose additional changes to the TSs to address the concerns discussed above within 30 days of the issuance of this amendment.

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 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 25, 1980

References

1. Letter dated October 1, 1976, from John F. Stolz (NRC) to Lowell E. Roe (TECo).

2. Letter dated December 7, 1976, from Lowell E. Roe to John F. Stolz.

3. Letter dated April 7, 1977, from Lowell E. Roe to John F. Stolz.

4. Letter dated March 20, 1978, from Lowell E. Roe to John F. Stolz.

5. Letter dated April 22, 1977, from Roger S. Boyd (NRC) to Lowell E. Roe.

6. Letter dated June 14, 1977, from John F. Stolz to Lowell E. Roe.

7. Letter dated June 24, 1977, from John F. Stolz to Lowell E. Roe.

8. NRC staff report "Reactor Vessel Pressure Transient Protection for Pressurized Water Reactors", NUREG-0224 dated September 1978.

9. Letter dated July 22, 1978, from Richard P. Crouse (TECo) to Robert W. Reid (NRC).