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JULY 25 1980

Docket No. 50-346

REGULATORY DOCKET FILE COPY

Mr. Richard P. Crouse
Vice President, Nuclear
Toledo Edison Company
Edison Plaza
300 Madison Avenue
Toledo, Ohio 43652

Dear Mr. Crouse:

The Commission has issued the enclosed Amendment* No. 28 to Facility Operating License No. NPF-3 for the Davis-Besse Nuclear Power Station, Unit No. 1. The amendment consists of changes to the Technical Specifications in response to your application dated March 20, 1978.

This amendment revises the Technical Specification setpoint for the automatic closure of the decay heat removal system isolation valves, and implements a pressurizer heater interlock which deenergizes the heaters if reactor coolant system pressurization is attempted and both decay heat removal system isolation valves are not closed. This action satisfies the requirements of license condition 2.C.(3)(d) and 2.C.(3)(j). In addition, the amendment adds a new condition to the license which requires you, prior to operation beyond five effective full power years, to provide to the NRC a reanalysis and proposed modifications, as necessary, to ensure continued means of protection against low temperature reactor coolant system overpressure events. License condition 2.C.(3)(d) is therefore replaced with the new requirement and condition 2.C.(3)(j) is therefore deleted.

You have committed to propose further changes concerning the overpressure protection system within 30 days from the date of this letter.

Copies of the Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

Original signed by
Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Licensing

8008050513 P

Enclosures:
1. Amendment No. 28
2. Safety Evaluation
3. Notice

CP

AD-O
TNov
7/2

*See previous yellow for concurrences.

OFFICE	cc w/enclosures:	See next page	ORB#4:DL	ORB#4:DL	C-ORB#4:DL	
SURNAME			RIngram	DGarner/cb*	~~RReid~~	MP
DATE			7/25/80	7/23/80	7/25/80	7

NRC FORM 318 (9-76) NRCM 0240

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Copies of the Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Licensing

Enclosures:
1. Amendment No.
2. Safety Evaluation
3. Notice

cc w/enclosures: See next page

Fed Reg + Amendment only

OFFICE	ORB#4:DL	ORB#4:DL	C-ORB#4:DL	AD-OR:DL	OELD	
SURNAME	RIngram	DGarner/cb	RReid	TNovak	M. Rothschild	
DATE	7/22/80	7/23/80	7/23/80	7/23/80	7/24/80	

ATTACHMENT TO LICENSE AMENDMENT NO. 28

FACILITY OPERATING LICENSE NO. NPF-3

DOCKET NO. 50-346

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages as indicated. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

Pages

3/4 3-11

3/4 3-12

3/4 3-13

3/4 3-22

3/4 4-2

3/4 5-4

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

THE TOLEDO EDISON COMPANY

AND

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

DOCKET NO. 50-346

DAVIS-BESSE NUCLEAR POWER STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 28
License No. NPF-3

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by the Toledo Edison Company and The Cleveland Electric Illuminating Company (the licensees) dated March 20, 1978, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, Facility Operating License No. NPF-3 is hereby amended as indicated below and by changes to the Technical Specifications as indicated in the attachment to this license amendment:

   A. Revise paragraph 2.C.(2) to read as follows:

      Technical Specifications

      The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 28, are hereby incorporated in the license. The Toledo Edison Company shall operate the facility in accordance with the Technical Specifications.

   B. Replace paragraph 2.C.(3)(d) with the following:

      Prior to operation beyond five Effective Full Power Years, the Toledo Edison Company shall provide to the NRC a renalysis and proposed modifications, as necessary, to ensure continued means of protection against low temperature reactor coolant system overpressure events.

   C. Delete paragraph 2.C.(3)(j).

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: July 25, 1980

cc w/enclosure(s):

Mr. Donald H. Hauser, Esq.
The Cleveland Electric
Illuminating Company
P. O. Box 5000
Cleveland, Ohio 44101

Gerald Charnoff, Esq.
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Harold Kahn, Staff Scientist
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361 East Broad Street
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Mr. Rick Jagger
Industrial Commission
State of Ohio
2323 West 5th Avenue
Columbus, Ohio 43216

Mr. Ted Myers
Licensing Engineer
Toledo Edison Company
Edison Plaza
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U. S. Nuclear Regulatory Commission
Resident Inspector's Office
5503 N. State Route 2
Oak Harbor, Ohio 43449

Director, Technical Assessment
Division
Office of Radiation Programs
(AW-459)
U. S. Environmental Protection Agency
Crystal Mall #2
Arlington, Virginia 20460

U. S. Environmental Protection Agency
Federal Activities Branch
Region V Office
ATTN: EIS COORDINATOR
230 South Dearborn Street
Chicago, Illinois 60604

cc w/enclosure(s) and incoming dtd.:
3/20/78

Ohio Department of Health
ATTN: Director of Health
450 East Town Street
Columbus, Ohio 43216

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

July 25, 1980

Docket No. 50-346

Mr. Richard P. Crouse
Vice President, Nuclear
Toledo Edison Company
Edison Plaza
300 Madison Avenue
Toledo, Ohio 43652

Dear Mr. Crouse:

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You have committed to propose further changes concerning the overpressure protection system within 30 days from the date of this letter.

Copies of the Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Licensing

Enclosures:
1. Amendment No. 28
2. Safety Evaluation
3. Notice

cc w/enclosures: See next page

TABLE 3.3-4

SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
INSTRUMENT STRINGS		
a. Containment Radiation	≤ 2 x Background at RATED THERMAL POWER	≤ 2 x Background at, RATED THERMAL POWER#
b. Containment Pressure - High	≤ 18.4 psia	≤ 18.52 psia#
c. Containment Pressure - High-High	≤ 38.4 psia	≤ 38.52 psia#
d. RCS Pressure - Low	≥ 1620.75 psig	≥ 1615.75 psig#
e. RCS Pressure - Low-Low	≥ 420.75 psig	≥ 415.75 psig#
f. BWST Level	≥ 49.5 and ≤ 55.0 in. H_2O	≥ 48.3 and ≤ 56.7 in. H_2O#
SEQUENCE LOGIC CHANNELS		
a. Essential Bus Feeder Breaker Trip (90%)	≥ 3744 volts for 7 ± 1.5 sec	≥ 3558 volts for 7 ± 1.5 sec#
b. Diesel Generator Start, Load Shed on Essential Bus (59%)	≥ 2071 and ≤ 2450 volts for 0.5 ± 0.1 sec	≥ 2071 and ≤ 2450 volts for 0.5 ± 0.1 sec#
INTERLOCK CHANNELS		
a. Decay Heat Isolation Valve and Pressurizer Heater	< 438 psig	< 443 psig#*

#Allowable Value for CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION.

* Referenced to the centerline of DH11 and DH12

TABLE 3.3-5

SAFETY FEATURES SYSTEM RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
1. Manual	
a. Fans	
1. Emergency Vent Fan	NA
2. Containment Cooler Fan	NA
b. HV & AC Isolation Valves	
1. ECCS Room	NA
2. Emergency Ventilation	NA
3. Containment Air Sample	NA
4. Containment Purge	NA
5. Pentration Room Purge	NA
c. Control Room HV & AC Units	NA
d. High Pressure Injection	
1. High Pressure Injection Pumps	NA
2. High Pressure Injection Valves	NA
e. Component Cooling Water	
1. Component Cooling Water Pumps	NA
2. Component Cooling Aux. Equip. Inlet Valves	NA
3. Component Cooling to Air Compressor Valves	NA
f. Service Water System	
1. Service Water Pumps	NA
2. Service Water From Component Cooling Heat Exchanger Isolation Valves	NA
g. Containment Spray Isolation Valves	NA
h. Emergency Diesel Generator	NA
i. Containment Isolation Valves	
1. Vacuum Relief	NA
2. Normal Sump	NA
3. RCS Letdown Delay Coil Outlet	NA
4. RCS Letdown High Temperature	NA

TABLE 3.3-3

SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF UNITS	UNITS TO TRIP	MINIMUM UNITS OPERABLE	APPLICABLE MODES	ACTION
3. MANUAL ACTUATION					
a. SFAS (except Containment Spray and Emergency Sump Recirculation)	2	2	2	ALL MODES	11
b. Containment Spray	2	2	2	1, 2, 3, 4	11
4. SEQUENCE LOGIC CHANNELS	4	2***	4	1, 2, 3, 4	9#
5. INTERLOCK CHANNELS					
a. Decay Heat Isolation Valve	1	1	1	1, 2, 3, 4, 5	12#
b. Pressurizer Heaters	2	2	2	3,4,5	13#

TABLE 3.3-3 (Continued)

TABLE NOTATION

* Trip function may be bypassed in this MODE with RCS pressure below 1800 psig. Bypass shall be automatically removed when RCS pressure exceeds 1800 psig.

** Trip function may be bypassed in this MODE with RCS pressure below 600 psig. Bypass shall be automatically removed when RCS pressure exceeds 600 psig.

*** One must be in SFAS Channels #1 or #3, the other must be in Channels #2 or #4.

\# The provisions of Specification 3.0.4 are not applicable.

ACTION STATEMENTS

ACTION 9 - With the number of OPERABLE functional units one less than the Total Number of Units operation may proceed provided both of the following conditions are satisfied:

a. The inoperable functional unit is placed in the tripped condition within one hour.

b. The Minimum Units OPERABLE requirement is met; however, one additional functional unit may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.1.

ACTION 10 - With any component in the Output Logic inoperable, trip the associated components within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 11 - With the number of OPERABLE Units one less than the Total Number of Units, restore the inoperable functional unit to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 12 - a. With less than the Mimimum Units OPERABLE and reactor coolant pressure $\geq$ 438 psig, both Decay Heat Isolation Valves (DH11 and DH12) shall be verified closed.

b. With Less than the Minimum Units OPERABLE and reactor coolant pressure < 438 psig operation may continue; however, the functional unit shall be OPERABLE prior to increasing reactor coolant pressure above 438 psig.

ACTION 13 - With less than the Minimum Units OPERABLE and reactor coolant pressure <438 psig, operation may continue; however, the functional unit shall be OPERABLE prior to increasing reactor coolant pressure above 438 psig, or the inoperable functional unit shall be placed in the tripped state.

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.

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.

ECCS SUBSYSTEMS - $T_{avg} > 280°F$

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent ECCS subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE high pressure injection (HPI) pump,

b. One OPERABLE low pressure injection (LPI) pump,

c. One OPERABLE decay heat cooler, and

d. An OPERABLE flow path capable of taking suction from the borated water storage tank (BWST) on a safety injection signal and automatically transferring suction to the containment sump on a borated water storage tank low level signal during the recirculation phase of operation.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one ECCS subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date.

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed or otherwise secured in position, is in its correct position.

SURVEILLANCE REQUIREMENTS

b. At least once per 31 days by verifying that the ECCS piping is full of water by venting the ECCS pump casings and discharge piping high points.

c. By a visual inspection which verifies that no loose debris (rags, trash, clothing, etc.) is present in the containment which could be transported to the containment emergency sump and cause restriction of the pump suction during LOCA conditions. This visual inspection shall be performed:

   1. For all accessible areas of the containment prior to establishing CONTAINMENT INTEGRITY, and

   2. Of the areas affected within containment at the completion of each containment entry when CONTAINMENT INTEGRITY is established.

d. At least once per 18 months by:

   1. Verifying that the interlocks:

      a) Close DH-11 and DH-12 and deenergize the pressurizer heaters, if either DH-11 or DH-12 is open and a simulated reactor coolant system pressure which is greater than the trip setpoint(<438 psig) is applied.
      b) Prevent the opening of DH-11 and DH-12 when a simulated or actual reactor coolant system pressure which is greater than the trip setpoint (<438 psig) is applied.

   2. A visual inspection of the containment emergency sump which verifies that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or corrosion.

   3. Verifying a total leak rate $\leq$ 20 gallons per hour for the LPI system at:

      a) Normal operating pressure or hydrostatic test pressure of $\geq$ 150 psig for those parts of the system downstream of the pump suction isolation valve, and

      b) $\geq$ 45 psig for the piping from the containment emergency sump isolation valve to the pump suction isolation valve.

   4. Verifying that a minimum of 72 cubic feet of solid granular trisodium phosphate dodecahydrate (TSP) is contained within the TSP storage baskets.

REACTOR COOLANT LOOPS

LIMITING CONDITION FOR OPERATION

3.4.1 Both reactor coolant loops and both reactor coolant pumps in each loop shall be in operation.

APPLICABILITY: As noted below, but excluding MODE 6.*

ACTION:

MODES 1 and 2:

a. With one reactor coolant pump not in operation, STARTUP and POWER OPERATION may be initiated and may proceed provided THERMAL POWER is restricted to less than 78.3% of RATED THERMAL POWER and within 4 hours the setpoints for the following trips have been reduced to the values specified in Specification 2.2.1 for operation with three reactor coolant pumps operating:

   1. High Flux
   2. Flux-ΔFlux-Flow

b. With one reactor coolant pump in each loop not in operation, STARTUP and POWER OPERATION may be initiated and may proceed provided THERMAL POWER is restricted to less than 50.6% of RATED THERMAL POWER and within 4 hours the setpoints for the following trips have been reduced to the values specified in Specification 2.2.1 for operation with one reactor coolant pump operating in each loop:

   1. High Flux
   2. Flux-ΔFlux-Flow

*See Special Test Exception 3.10.3.

JUN 7 1979

LIMITING CONDITION FOR OPERATION (Continued)

MODES 3, 4 and 5:

a. Operation may proceed provided at least one reactor coolant loop is in operation with an associated reactor coolant pump or decay heat removal pump.*

b. Not more than one decay heat removal pump may be operated with the sole suction path through DH-11 and DH-12 unless the control power has been removed from the DH-11 and DH-12 valve operators, or manual valves DH-21 and DH-23 are opened.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

*All reactor coolant pumps and decay heat removal train pumps may be de-energized for up to 1 hour to accommodate surveillance testing & pre-operational testing, provided no operations are permitted which could cause dilution of the reactor coolant system boron concentration.

SURVEILLANCE REQUIREMENTS

4.4.1 The Reactor Protective Instrumentation channels specified in the applicable ACTION statement above shall be verified to have had their trip setpoints changed to the values specified in Specification 2.2.1 for the applicable number of reactor coolant pumps operating either:

a. Within 4 hours after switching to a different pump combination if the switch is made while operating, or

b. Prior to reactor criticality if the switch is made while shutdown.

TABLE 4.3-2

SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. INSTRUMENT STRINGS				
a. Containment Radiation - High	S	R	M	All MODES
b. Containment Pressure - High	S	R	M(2)	1, 2, 3
c. Containment Pressure - High-High	S	R	M(2)	1, 2, 3
d. RCS Pressure - Low	S	R	M	1, 2, 3
e. RCS Pressure - Low-Low	S	R	M	1, 2, 3
f. BWST Level - Low	S	R	M	1, 2, 3
2. OUTPUT LOGIC				
a. Incident Level #1: Containment Isolation	S	R	M	All MODES
b. Incident Level #2: High Pressure Injection and Starting Diesel Generators	S	R	M	1, 2, 3, 4
c. Incident Level #3: Low Pressure Injection	S	R	M	1, 2, 3, 4
d. Incident Level #4: Containment Spray	S	R	M	1, 2, 3, 4
e. Incident Level #5: Containment Sump Recirculation	S	R	M	1, 2, 3, 4
3. MANUAL ACTUATION				
a. SFAS (Except Containment Spray and Emergency Sump Recirculation)	NA	NA	M(1)	All MODES
b. Containment Spray	NA	NA	M(1)	1, 2, 3
4. SEQUENCE LOGIC CHANNELS	S	NA	M	1, 2, 3, 4

TABLE 4.3-2

SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
5. INTERLOCK CHANNELS				
a. Decay Heat Isolation Valve	S	R	**	1, 2, 3, 4, 5
b. Pressurizer Heater	S	R	**	3,4,5

**See Specification 4.5.2.d.1

TABLE NOTATION

(1) Manual actuation switches shall be tested at least once per 18 months during shutdown. All other circuitry associated with manual safeguards actuation shall receive a CHANNEL FUNCTIONAL TEST at least once per 31 days.

(2) The CHANNEL FUNCTIONAL TEST shall include exercising the transmitter by applying either vacuum or pressure to the appropriate side of the transmitter.

> 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).

7590-01

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-346

THE TOLEDO EDISON COMPANY

AND

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 28 to Facility Operating License No. NPF-3, issued to The Toledo Edison Company and The Cleveland Electric Illuminating Company (the licensees), which revised the license and Technical Specifications for operation of the Davis-Besse Nuclear Power Station, Unit No. 1 (the facility) located in Ottawa County, Ohio. The amendment is effective as of its date of issuance.

This amendment revises the Technical Specification setpoint for the automatic closure of the decay heat removal system isolation valves, and implements a pressurizer heater interlock which deenergizes the heaters if reactor coolant system pressurization is attempted and both decay heat removal system isolation valves are not closed. This action satisfies the requirements of license conditions 2.C.(3)(d) and 2.C.(3)(j). In addition, the amendment adds a new condition to the license which requires, prior to operation beyond five effective full power years, the Toledo Edison Company to provide a reanalysis and proposed modifications, as necessary, to ensure continued means of protection against low temperature reactor coolant system overpressure events. Therefore, license condition 2.C.(3)(d) is replaced with the above new requirement and condition 2.C.(3)(j) is removed from the license.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

For further details with respect to this action, see (1) the application for amendment dated March 20, 1978, (2) Amendment No. 28 to License No. NPF-3, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C., and at the Ida Rupp Public Library, 310 Madison Street, Port Clinton, Ohio. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 25th day of July 1980.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Licensing