ATTACHMENT TO LICENSE AMENDMENT NO. 6

FACILITY OPERATING LICENSE NO. DPR-73

DOCKET NO. 50-320

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

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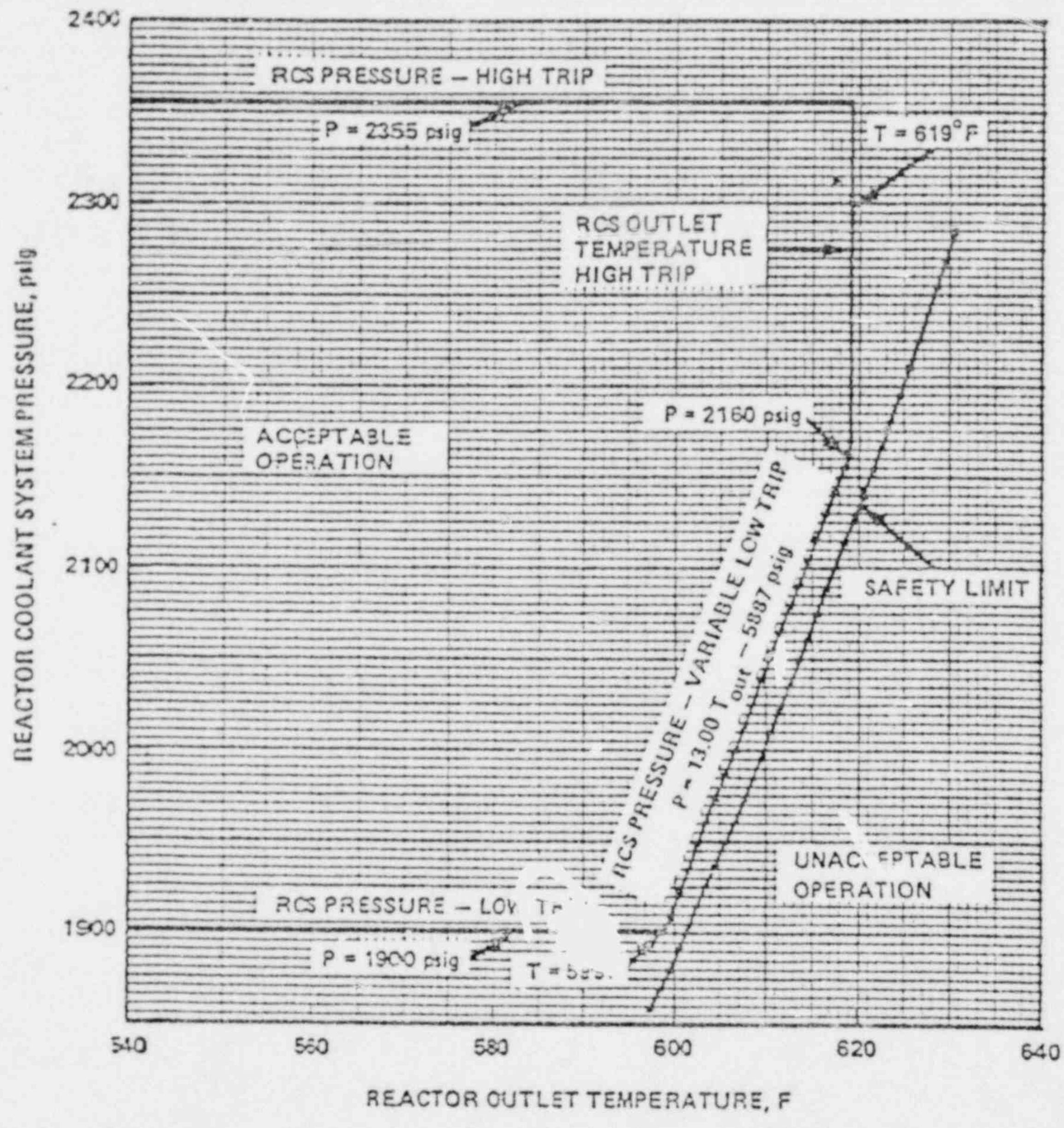

Figure 2.1—1 Reactor Core Safety Limit

TABLE 2.2-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Nuclear Overpower	$\leq$ 105.5% of RATED THERMAL POWER with four pumps operating	$\leq$ 105.6% of RATED THERMAL POWER with four pumps operating#
	$\leq$ 78.1% of RATED THERMAL POWER with three pumps operating	$\leq$ 78.2% of RATED THERMAL POWER with three pumps operating#
	$\leq$ 50.9% of RATED THERMAL POWER with one pump operating in each loop	$\leq$ 51.0% of RATED THERMAL POWER with one pump operating in each loop#
3. RCS Outlet Temperature-High	$\leq$ 619°F	$\leq$ 619.08°F#
4. Nuclear Overpower Based on RCS Flow and AXIAL POWER IMBLANCE (1)	Trip Setpoint not to exceed the limit line of Figure 2.2-1.	Allowable Values not to exceed the limit line of Figure 2.2-2.#
5. RCS Pressure-Low (1)	$\geq$ 1900 psig	$\geq$ 1899.0 psig*; $\geq$ 1891.5 psig **
6. RCS Pressure-High	$\leq$ 2355 psig	$\leq$ 2356.0 psig*; $\leq$ 2363.5 psig **
7. RCS Pressure-Variable Low (1)	$\geq$ (13.00 T_{out}°F - 5887) psig	$\geq$ (13.00 T_{out}°F - 5887.64) psig#

TABLE 2.2-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTION UNIT	TRIP SETPOINT	ALLOWABLE VALUES
8. Nuclear Overpower based on Pump Monitors[(1)]	≤ 125% of RATED THERMAL POWER with three pumps operating	≤ 125% of RATED THERMAL POWER with three pumps operating#
	≤ 56.9% of RATED THERMAL POWER with one pump operating in each loop	≤ 57.18% of RATED THERMAL POWER with one pump operating in each loo
	≤ 0% of RATED THERMAL POWER with two pump operating in one loop and no pump operating in the other loop	≤ 0.28% of RATED THERMAL POWER w two pumps operating in one loop and no pump operating in the other loop
	≤ 0% of RATED THERMAL POWER with no pumps operating or only one pump operating	≤ 0.28% of RATED THERMAL POWER with no pumps operating or only one pump operating#
9. Reactor Containment Vessel	≤ 4 psig	≤ 4 psig#

(1)Trip may be manually bypassed when RCS pressure ≤ 1720 psig by actuating Shutdown Bypass provided that:

a. The Nuclear Overpower Trip Setpoint is ≤ 5% of RATED THERMAL POWER
b. The Shutdown Bypass RCS Pressure - High Trip Setpoint of ≤ 1720 psig is imposed, and
c. The Shutdown Bypass is removed when RCS Pressure > 1800 psig.

*Allowable value for Channel Functional Test.
**Allowable value for Channel Calibration.
#Allowable value for Channel Functional Test and Channel Calibration.

2.2.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

The Reactor Protection System Instrumentation Trip Setpoint specified in Table 2.2-1 are the values at which the Reactor Trips are set for each parameter. The Trip Setpoints have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their safety limits. Operation with a trip setpoint less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that each Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analyses.

The Shutdown Bypass provides for bypassing certain functions of the Reactor Protection System in order to permit control rod drive tests, zero power PHYSICS TESTS and certain startup and shutdown procedures. The purpose of the Shutdown Bypass RCS Pressure-High trip is to prevent normal operation with Shutdown Bypass activated. This high pressure trip setpoint is lower than the normal low pressure trip setpoint so that the reactor must be tripped before the bypass is initiated. The Nuclear Overpower Trip Setpoint of $\leq$ 5.0% prevents any significant reactor power from being produced. Sufficient natural circulation would be available to remove 5.0% of RATED THERMAL POWER if none of the reactor coolant pumps were operating.

Manual Reactor Trip

The Manual Reactor Trip is a redundant channel to the automatic Reactor Protection System instrumentation channels and provides manual reactor trip capability.

Nuclear Overpower

A Nuclear Overpower trip at high power level (neutron flux) provides reactor core protection against reactivity excursions which are too rapid to be protected by temperature and pressure protective circuitry.

During normal station operation, reactor trip is initiated when the reactor power level reaches 105.5% of rated power. Due to calibration and instrument errors, the maximum actual power at which a trip would be actuated could be 112%, which was used in the safety analysis.

The AXIAL POWER IMBALANCE boundaries are established in order to prevent reactor thermal limits from being exceeded. These thermal limits are either power peaking kw/ft limits or DNBR limits. The AXIAL POWER IMBALANCE reduces the power level trip produced by the flux-to-flow ratio such that the boundaries of Figure 2.2-1 are produced. The flux-to-flow ratio reduces the power level trip and associated reactor power-reactor power-imbalance boundaries by 1.05% for a 1% flow reduction.

RCS Pressure - Low, High and Variable Low

The High and Low trips are provided to limit the pressure range in which reactor operation is permitted.

During a slow reactivity insertion startup accident from low power or a slow reactivity insertion from high power, the RCS Pressure-High setpoint is reached before the Nuclear Overpower Trip Setpoint. The trip setpoint for RCS Pressure-High, 2355 psig, has been established to maintain the system pressure below the safety limit, 2750 psig, for any design transient. The RCS Pressure-High trip is backed up by the pressurizer code safety valves for RCS over pressure protection, and is therefore set lower than the set pressure for these valves, 2500 psig. The RCS Pressure-High trip also backs up the Nuclear Overpower trip.

The RCS Pressure-Low, 1800 psig, and RCS Pressure-Variable Low, (13.00 T_{out} °F-5887) psig, Trip Setpoints have been established to maintain the DNB ratio greater than or equal to 1.30 for those design accidents that result in a pressure reduction. It also prevents reactor operation at pressures below the valid range of DNB correlation limits, protecting against DNB.

Due to the calibration and instrumentation errors, the safety analysis used a RCS Pressure-Variable Low Trip Setpoint of (13.00 T_{out} °F-5927) psig.

Nuclear Overpower Based on Pump Monitors

In conjunction with the power/imbalance/flow trips the Nuclear Overpower Based On Pump Monitors trip prevents the minimum core DNBR from decreasing below 1.30 by tripping the reactor due to the loss of reactor coolant pump(s). The pump monitors also restrict the power level for the number of pumps in operation.

TABLE 3.3-1 (Continued)

TABLE NOTATION

*With the control rod drive trip breakers in the closed position and the control rod drive system capable of rod withdrawal.

**When Shutdown Bypass is actuated.

#The provisions of Specification 3.0.4 are not applicable.

##High voltage to detector may be de-energized above 10^{-10} amps on both Intermediate Range channels.

(a) Trip may be manually bypassed when RCS pressure ≤ 1820 psig by actuating Shutdown Bypass provided that:

(1) The Nuclear Overpower Trip Setpoint is ≤ 5% of RATED THERMAL POWER.

(2) The Shutdown Bypass RCS Pressure--High Trip Setpoint of ≤ 1820 psig is imposed.

(3) The Shutdown Bypass is removed when RCS pressure > 1900 psig.

(b) Trip may be bypassed during testing pursuant to Special Test Exception 3.10.3.

ACTION STATEMENTS

ACTION 1 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and/or open the control rod drive trip breakers.

ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels STARTUP and/or POWER OPERATION may proceed provided all of the following conditions are satisfied:

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

b. The Minimum Channels OPERABLE requirement is met; however, one additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1.1,

TABLE 3.3-3 (Continued)

TABLE NOTATION

* Trip function may be bypassed in this MODE with RCS pressure below 1920 psig. Bypass shall be automatically removed when RCS pressure exceeds 1950 psig.

** 3 channels per Automatic Actuation Logic, Each R. B. Pressure High Channel trips one Safety Injection Channel and one R. B. Cooling & Isolation Channel.

*** 3 channels per Automatic Actuation Logic, R. B. Spray Valves are actuated by R. B. Cooling and Isolation.

**** Trip function may be bypassed in this mode with steam generator pressure $\leq$ 800 psig. Bypass shall be removed when steam generator pressure $\geq$ 800 psig.

\# The provisions of Specification 3.0.4 are not applicable.

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

AUG 17 1978

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
SUPPORTING AMENDMENT NO. 6 TO FACILITY OPERATING LICENSE NO. DPR-73

METROPOLITAN EDISON COMPANY
JERSEY CENTRAL POWER & LIGHT COMPANY
PENNSYLVANIA ELECTRIC COMPANY

DOCKET NO. 50-320

THREE MILE ISLAND NUCLEAR STATION, UNIT 2

1. Containment Air Lock Seal Leak Rate Testing

Introduction

By letter dated May 19, 1978 transmitting Technical Specification Change Request No. 009, Metropolitan Edison Company (Met Ed) requested amendment of Appendix A to Facility Operating License No. DPR-73 for Three Mile Island Nuclear Station, Unit 2 (TMI-2). The requested change would amend the Technical Specifications to permit a more effective method of seal leakage verification.

Discussion

The present wording of the TMI-2 Technical Specifications requires that containment air lock seal leak rate testing be performed "by pressure decay when the volume between the door seals is pressurized to $\geq$ 10 psig..." The acceptance criteria specified, (Leakage $\leq$ 0.01 La) translates to a pressure drop of 10 psig in a period of 10 seconds. This is inconsistent with the additional requirement to maintain door seals pressurized to $\geq$ 10 psig for at least 15 minutes. In addition because the manufacturer has indicated that the volume between the door seals should not be pressurized above 10 psig, and because the volume between the door seals is quite small ($\sim$ 0.02 cu. ft.), Met Ed states that it is not possible to perform the surveillance using the pressure decay method. The proposed wording of the Technical Specifications would allow measurement of leak rate testing by another method (e.g., the flow monitoring method). This proposed change deletes the requirement to measure seal leakage by a pressure drop test method, and specifies the pressure at which the seal leak rate is to be determined using a flow meter.

The basis of the surveillance requirement is to provide assurance that the containment leakage rates of Limiting Condition for Operation 3.6.1.2 are not exceeded as a result of seal damage occurring during door usage.

Dupe of

The proposed. Technical Specification Change does not change the limit established for allowable leakage through the door seals. This limit remains $\leq$ 0.01 La (1 percent of the total allowable containment leak rate).

Evaluation

We have reviewed the information provided by the licensee as well as additional information from our Office of Inspection and Enforcement and find that the proposed method of seal leakage measurement is effective (in fact, for this particular application, it is more effective than that required by the present Technical Specifications), satisfies the intent and the basis of the Technical Specification, and therefore provides equal or greater assurance that seal leakage will be within acceptable limits.

Based on the above, we conclude that the proposed change permitting an alternate method of measuring containment air lock seal leakage is acceptable, and that the facility operating license can be amended by changing the Technical Specifications as shown in the attachment to this license amendment.

2. Ultimate Heat Sink Temperature

Introduction

By letter dated June 5, 1978 transmitting Technical Specification Change Request No. 011, Metropolitan Edison Company (Met Ed) requested amendment of Appendix A to Facility Operating License No. DPR-73 for Three Mile Island Nuclear Station, Unit 2 (TMI-2). The requested change would amend the Technical Specifications to permit plant operation with temperature of the Susquehanna River (the ultimate heat sink) in excess of the present limit of 85°F.

Discussion

The present Technical Specifications require shutdown of TMI-2 within 30 hours if heat sink (Susquehanna River) temperature exceeds 85°F. Past operating experience shows that river water temperature may exceed 85°F during the summer months.

The licensee has stated that with one exception, sufficient margin exists in the design of all safety-related equipment which would reject heat to the ultimate heat sink, such that the equipment will operate acceptably with the heat sink temperature up to 95°F. The exception is the control building air conditioning equipment, which can operate satisfactorily with heat sink temperatures up to 90°F. With increased flow, which will be available after replacement of the control building booster pump impellers, this equipment will also operate satisfactorily at heat sink temperatures up to 95°F.

The licensee has provided results of their analyses and additional information to confirm that each safetyrelated component will operate within its design parameters and will not suffer degradation of performance or impairment in performance of its safety function for the proposed increased heat sink temperatures.

Evaluation

Based on our evaluation of the results of the licensee's analyses of component performance and on our calculations and estimates of system performance based on component design temperatures, we find reasonable assurance that safety-related components will operate within their design parameters and will not suffer impairment in performance of their safety functions at the proposed increased ultimate heat sink temperatures. We therefore find that operation at the proposed heat sink temperatures will not cause a significant decrease in the performance margins of safety-related systems, and that such operation is acceptable.

Based on the above, we conclude that the facility operating license can be amended by changing the Technical Specifications as shown in the attachment to this license amendment.

3. Orifice Rod and Burnable Poison Rod Assemblies

Introduction

By letter dated July 7, 1978 transmitting Technical Specification Change Request No. 014, Metropolitan Edison Company (Met Ed) requested amendment of Appendix A to Facility Operating License No. DPR-73 for Three Mile Island Nuclear Station, Unit 2 (TMI-2). The requested changes would amend the Technical Specifications to permit removal of all but two orifice rod assemblies (ORA's) and installation of retainers on the remaining two ORA's and on the burnable poison rod assemblies (BPRA's). These changes are proposed because of the concern over wear of the fuel assembly holddown latch assemblies as found in other plants, caused by levitation and vibration of the ORA's and BPRA's.

Additional changes covered by this change request are the following, which which are not related to ORA removal or BPRA installation:

- Increase in RCS pressure - low trip setpoint by 100 psig and corresponding increase in the high pressure trip during startup in shutdown bypass.

- Correction of rod bow penalty to correctly reflect the NRC rod bow model.

- Addition of allowable values for Channel Functional Test to account for instrument errors.

Discussion

ORA's had been provided in guide tubes not containing control rod assemblies or axial power shaping rod assemblies to limit reactor coolant bypass flow through otherwise empty guide tubes. BPRA's are used to provide partial control of slowly ocurring negative reactivity changes and to flatten the radial power distribution.

A burnable poison rod assembly (BPRA) was ejected from the core at one of B&W's Mark B plants. B&W analyzed the problem and determined it to be caused by levitation of the BPRAs during four-pump operation and subsequent fretting wear in the holddown latching mechanism. After the initial inspection of fuel assemblies at the affected plant, B&W also observed visual indications of wear at other plants in the idential latching mechanisms of BPRAs, assemblies that held orifice rods (ORAs), and source or modified orifice rod assemblies (MORAs). To resolve this problem in TMI-2, the licensee proposes installation of retainers on the BPRA's and on two modified ORA's, and removal of the remaining thirty-eight ORA's. Information supporting this proposal, attached or referenced in the submittal of July 17, 1978, includes:

- B&W letter to NRC dated June 7, 1978, Taylor to Varga
- BAW-1496, "BPRA Retainer Design Report," May 1978
- BAW-1497, "Justification for Removal of Orifice Rod Assemblies in Three Mile Island Unit 2, Cycle 1," June , 1978.

The submittal states that installation of the retainers would reduce reactor coolant system (RCS) flow by less than 1 percent and removal of the ORA's would increase bypass flow in the hot assembly by 1.6 percent. To compensate for core flow distribution effects caused by the changes, the licensee proposed increasing the primary system flow rate (flow requirement increase of 2 percent). The present margin in flow rate between measured and technical specification requirement (5 percent) would be reduced. Because this operating margin is reduced from 5 percent to 3 percent, flow instrumentation was evaluated to assure that its accuracy is within the range of the margin. The flow measurement system and its calibration were identified by the licensee to be identical to the system for Three Mile Island, Unit 1 which has previously been shown to have a measurement uncertainty of about 1.5 percent. This uncertainty is within the 3 percent margin available.

The limiting fuel assembly does not contain a BPRA during cycle 1 operation. Though this would further increase flow in the hot assembly, no credit was taken for it. The net effect of the increased flow and bypass penalties is a slight increase in DNBR's.

DNBR-limited transients were reanalyzed considering the increased flow, trip setting adjustments, uncertainties, and rod bow penalties (for cycle 1 a DNBR criterion of 1.41 accounts for rod bow effects). DNBR values of 1.65 for the 4-pump coastdown event and 1.58 for a feedwater temperature decrease event were calculated. The 1-pump coast-down from 4-pump operation was identified to be the most limiting flow transient because it is used to determine the flux/flow trip set point. Discussion with B&W indicates that the DNBR for the 1-pump coastdown is 1.43.

A retainer device has been designed and tested by B&W to ensure positive holddown of BPRAs, ORAs and MORAs during reactor operation. The design and the test results were reported to NRC in BAW-1496 and the above-referenced letter of June 7, 1978. For continued operation of TMI-2, Metropolitan Edison Company proposes to install the retainer devices on 68 BPRAs and 2 MORAs. All regular ORAs (38) will be removed from the core. These changes apply only for the remainder of the current cycle, Cycle 1, at which time BPRAs are usually withdrawn from the core.

The potential consequences of a retainer failure have also been addressed although failure is considered unlikely. The neutronic and thermal-hydraulic consequences are considered small. Interference with control rod motion, for example, would not, according to analysis of stuck-out control rod transients for B&W 177-FA plants, prevent safe shutdown of the plant.

The major concern associated with retainer failure is plant damage, primarily in the steam generators, and potential outages for repair. This damage should be precluded by the Loose Parts Monitoring System (LPMS). The LPMS is designed to detect a failed retainer in either the reactor vessel or steam generator. Even though the retainer device is designed for only one cycle of operation, B&W has stated that it will recommend that surveillance inspections be made following retainer use. This should provide additional confirmation of acceptable operation. B&W has also stated that definite plans regarding surveillance will be provided to NRC as they are formulated.

The first of the additional changes is the increase in RCS Pressure - low trip setpoint from 1800 psig to 1900 psig. This change is being made for greater operating flexibility and to increase the margin to high pressure injection (HPI) so that a rapid depressurization will not unnecessarily cause HPI as frequently as would occur with less margin. As a result of this increase in the RCS pressure - low trip setpoint, it is correspondingly necessary to increase the manual bypass by 100 psi to $\geq$ 1820 psig to incorporate 1820 psig as the new high pressure trip during startup in shutdown bypass. This will enable startup to be performed more easily and will continue to maintain the same margin previously used to allow for instrument errors.

The rod bow penalty has been revised to correctly reflect the NRC rod bow model. The original Technical Specification was prepared using the B&W rod bow model and during investigation into the removal of the ORAs was discovered and is corrected herein.

Also incorporated in this change is the addition of the allowable values for the Channel Functional Test which previously were not included in the Technical Specifications. These values have been added to account for instrument calibration error, instrument drift and instrument error.

Evaluation

We have reviewed the effect on core flow of installation of retainers on the MORA's and BPRA's and on core bypass flow of removal of the ORA's. Based on our review of the submitted data and on our calculations on similar changes previously approved for Davis Besse Unit 1, we find that the calculated reduction in core flow of 1 percent and increase in bypass flow of 1.6 percent are reasonable and acceptable.

Based on the similarity of flow instrumentation to that on TMI-1, and our previous evaluation of the flow measurement uncertainty for TMI-1, we find that Unit 2 flow measurement instrumentation accuracy is expected to be within the 3 percent operating margin between measured and technical specification flow rates. We have reviewed the adequacy of the additional 2 percent RCS flow to compensate for the 1.6 percent increase in core bypass introduced by the core modifications. It is estimated that the RCS flow increase would provide an additional 1.8 percent RCS flow through the core, which is greater than the 1.6 percent reduction because of the ORA bypass. Since there is no significant reduction in safety margins, we find the proposed core modifications acceptable.

We have reviewed the DNBR evaluations for the 4-pump coastdown, the feedwater temperature decrease event, and the one-pump coastdown from 4-pump operation, and find that the results for these limiting transients are above the cycle 1 DNBR criterion of 1.41, and are acceptable.

With regard to the ORA and BPRA retainers, based on (1) design analyses and test results on the retainer device, (2) analyses which indicate that failure of the retainers, however unlikely, would not prevent shutdown and (3) failure detection capability of the Loose Parts Monitoring System, we find that there is reasonable assurance that the retainers will provide adequate holddown force on the BPRAs and MORAs and that the proposed use of the retainer devices in TMI-2 is acceptable.

We have reviewed the proposed increase in the RCS pressure-low trip set point and the associated increase in the high pressure trip during startup, and find that since these increases result in the same or larger safety margins, they are acceptable.

The original calculations for a rod bowing penalty had been performed with a B&W rod bow model that we found unacceptable. We have verified that the revised rod bow model as presented in the change request conforms with the NRC-approved rod bow equation for B&W plants. Therefore, we find this change acceptable.

The allowable values for channel functional test in Technical Specification Table 2.2-1 reasonably account for various instrument errors, and we therefore find these changes acceptable.

In summary, we have evaluated the proposed changes in Technical Specification Change Request No. 014, and having found them all acceptable, we conclude that the facility operating license can be amended by changing the Technical Specifications as shown in the attachment to this license amendment.

4. Main Steam Safety Valves

Introduction

By letter dated July 24, 1978 transmitting Technical Specification Change Request No. 015, Metropolitan Edison Company (Met Ed) requested amendment of Appendix A to Facility Operating License No. DPR-73 for Three Mile Island Nuclear Station, Unit 2 (TMI-2). The requested change would amend the Technical Specifications to permit replacement of the original 12 dual discharge port main steam safety valves with 20 somewhat smaller single discharge port valves.

Discussion

During a previous event at TMI-2, some of the original main steam safety valves failed to close at an appropriate pressure after actuation. Efforts to modify the valves to eliminate the problem were unsuccessful, and the licensee elected to replace the valves. These Technical Specifications changes are required to reflect this design change.

The new safety valves provide a relief capacity of 120 percent of the total secondary steam flow compared with 114 percent provided by the original valves. The licensee states that all system modifications conform with requirements of appropriate sections of the ASME Code and with criteria previously accepted in the Final Safety Analysis Report (FSAR).

Evaluation

We have evaluated the information provided by the licensee and find that since the relieving capacity of the new main steam safety valves exceeds that originally provided, the proposed change is in the conservative direction and is therefore acceptable.

We further find acceptable the licensee's statements regarding conformance of all modifications with ASME Code and FSAR criteria.

Among the changes proposed by the licensee is a revision of the equation on page B 3/4 7-1 of the Technical Specifications for determining reduced Nuclear Overpower Trip Setpoint for inoperable safety valves. The proposed equation is essentially identical numerically to the original, and we do not find sufficient justification to make the proposed change.

Based on the above, we conclude that the proposed changes in the Technical Specifications covering the new main steam safety valves are acceptable, except as noted above, and that the facility operating license can be amended by changing the Technical Specifications as shown in the attachment to this license 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.

H. Silver, Project Manager
Light Water Reactors Branch No. 4
Division of Project Management

Steven A. Varga, Chief
Light Water Reactors Branch No. 4
Division of Project Management

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

Reference 41

AUG 1 7 1978

Docket No: 50-320

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Metropolitan Edison Company
ATTN: Mr. John G. Herbein
Vice President
P. O. Box 542
Reading, Pennsylvania 19603

Gentlemen:

SUBJECT: THREE MILE ISLAND NUCLEAR STATION, UNIT 2 - ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 6 to Facility Operating License No. DPR-73 which is effective as of the date of issuance.

Amendment No. 6 is in response to the following Technical Specification Change Requests to amend Appendix A of Facility Operating License DPR-73:

Change Request No.	Date
009	May 19, 1978
011	June 5, 1978
014	July 7, 1978
015	July 24, 1978

The amendment consists of the following:

1. Changes in license Paragraph 2.C.(2), and in Appendix A, Technical Specifications.

We have determined that Amendment No. 6 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, negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

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COPY SENT REGION I

Copies of the FEDERAL REGISTER Notice of Issuance and the safety evaluation supporting Amendment No. 6 are also enclosed.

Sincerely,

Steven A. Varga

Steven A. Varga, Chief
Light Water Reactors Branch 4
Division of Project Management

Enclosures:
1. Amendment No. 6
2. Federal Register Notice
3. Safety Evaluation

cc: w/encl.
See next page