July 23, 1991

Docket No. 50-265

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Mr. Thomas J. Kovach
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Dear Mr. Kovach:

SUBJECT: ISSUANCE OF AMENDMENTS (TAC NO. 80299)

The Commission has issued the enclosed Amendment No. 125 to Facility Operating License No. DPR-30 for the Quad Cities Nuclear Power Station, Unit 2. The amendment is in response to your application dated April 18, 1991.

The amendment changes the Technical Specifications to reflect a modification to the fast acting solenoid valves which initiate rapid closure of the turbine control valves. The new design uses a pressure switch, rather than a limit switch, to initiate a reactor scram. This amendment becomes effective as of the startup from refueling outage No. 11.

A copy of the related Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Original Signed By:

Leonard N. Olshan, Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 125 to DPR-30
2. Safety Evaluation

cc w/enclosures:
See next page

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Surname:	CMoore	LOlshan:ta	RBarrett	RBachmann
Date:	7/12/91	7/2/91	7/2/91	7/10/91

DFol
11

Mr. Thomas J. Kovach
Commonwealth Edison Company

Quad Cities Nuclear Power Station
Unit Nos. 1 and 2

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

COMMONWEALTH EDISON COMPANY

AND

IOWA-ILLINOIS GAS AND ELECTRIC COMPANY

DOCKET NO. 50-265

QUAD CITIES NUCLEAR POWER STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 125
License No. DPR-30

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

   A. The application for amendment by Commonwealth Edison Company (the licensee) dated April 18, 1991, 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, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B. of Facility Operating License No. DPR-30 is hereby amended to read as follows:

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B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 125 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the startup from refueling outage No. 11.

FOR THE NUCLEAR REGULATORY COMMISSION

Byron L. Siegel for

Richard J. Barrett, Director
Project Directorate III-2
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: July 23, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 125

FACILITY OPERATING LICENSE NO. DPR-30

DOCKET NO. 50-265

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change.

REMOVE	INSERT
1.1/2.1-2a	1.1/2.1-2a
1.1/2.1-3	1.1/2.1-3
1.1/2.1-10	1.1/2.1-10
1.1/2.1-11	1.1/2.1-11
	1.1/2.1-12
3.1/4.1-5	3.1/4.1-5
3.1/4.1-6	3.1/4.1-6
3.1/4.1-7	3.1/4.1-7
	3.1/4.1-7a
	3.1/4.1-7b
3.1/4.1-10	3.1/4.1-10
3.1/4.1-11	3.1/4.1-11
	3.1/4.1-11a
3.1/4.1-14	3.1/4.1-14
	3.1/4.1-15

The definitions used above for the APRM scram trip apply. In the event of operation with a maximum fraction limiting power density (MFLPD) greater than the fraction of rated power (FRP), the setting shall be modified as follows:

$$S \leq (.58W_D + 50) \frac{FRP}{MFLPD}$$

The definitions used above for the APRM scram trip apply.

The ratio of FRP to MFLPD shall be set equal to 1.0 unless the actual operating value is less than 1.0, in which case the actual operating value will be used.

This adjustment may also be performed by increasing the APRM gain by the inverse ratio, MFLPD/FRP, which accomplishes the same degree of protection as reducing the trip setting by FRP/MFLPD.

C. Reactor low water level scram setting shall be 144 inches above the top of the active fuel* at normal operating conditions.

D. Reactor low water level ECCS initiation shall be $\geq$ 84 inches above the top of the active fuel* at normal operating conditions.

E. Turbine stop valve scram shall be $\leq$ 10% valve closure from full open.

F. The scram for turbine control valve fast closure due to actuation of the fast acting solenoid valve shall be $\geq$ 460 psig EHC fluid pressure.

*Top of active fuel is defined to be 360 inches above vessel zero (See Bases 3.2).

G. Main steamline isolation valve closure scram shall be $\leq$ 10% valve closure from full open.

H. Main steamline low-pressure initiation of main steamline isolation valve closure shall be $\geq$ 825 psig.

I. Turbine EHC control fluid low-pressure scram on loss of control oil pressure shall be set at greater than or equal to 900 psig.

J. Condenser low vacuum scram shall be set at $\geq$ 21 inches Hg vacuum.

The trip setpoint of $\geq$ 460 psig EHC fluid pressure was developed to ensure that the pressure switch is actuated prior to the closure of the turbine control valves (at approximately 400 psig EHC fluid pressure) yet assure that the system is not actuated unnecessarily due to EHC system pressure transients which may cause EHC system pressure to momentarily decrease.

G. Reactor Coolant Low Pressure Initiates Main Steam Isolation Valve Closure

The low-pressure isolation at 825 psig was provided to give protection against fast reactor depressurization and the resulting rapid cooldown of the vessel. Advantage was taken of the scram feature which occurs in the Run mode when the main steamline isolation valves are closed to provide for reactor shutdown so that operation at pressures lower than those specified in the thermal hydraulic safety limit does not occur, although operation at a pressure lower than 825 psig would not necessarily constitute an unsafe condition.

H. Main Steamline Isolation Valve Closure Scram

The low-pressure isolation of the main steamlines at 825 psig was provided to give protection against rapid reactor depressurization and the resulting rapid cooldown of the vessel. Advantage was taken of the scram feature in the Run mode which occurs when the main steamline isolation valves are closed to provide for reactor shutdown so that high power operation at low reactor pressures does not occur, thus providing protection for the fuel cladding integrity safety limit. Operation of the reactor at pressures lower than 825 psig requires that the reactor mode switch be in the Startup position, where protection of the fuel cladding integrity safety limit is provided by the IRM and APRM high neutron flux scrams. Thus, the combination of main steamline low-pressure isolation and isolation valve closure scram in the Run mode assures the availability of neutron flux scram protection over the entire range of applicability of the fuel cladding integrity safety limit. In addition, the isolation valve closure scram in the Run mode anticipates the pressure and flux transients which occur during normal or inadvertent isolation valve closure. With the scrams set at 10% valve closure in the Run mode, there is no increase in neutron flux.

I. Turbine EHC Control Fluid Low-Pressure Scram

The turbine EHC control system operates using high-pressure oil. There are several points in this oil system where a loss of oil pressure could result in a fast closure of the turbine control valves. This fast closure of the turbine control valves is not protected by the turbine control valve fast closure scram, since failure of the oil system would not result in the fast closure solenoid valves being actuated. For a turbine control valve fast closure, the core would be protected by the APRM and high-reactor pressure scrams. However, to provide the same margins as provided for the generator load rejection on fast closure of the turbine control valves, a scram has been added to the reactor protection system which senses failure of control oil pressure to the turbine control system. This is an anticipatory scram and results in reactor shutdown before any significant increase in neutron flux occurs. The transient response is very similar to that resulting from the turbine control valve fast closure scram. The scram setpoint of 900 psig is set high enough to provide the necessary anticipatory function and low enough to minimize the number of spurious scrams. Normal operating pressure for this system is 1250 psig. Finally, the control valves will not start until the fluid pressure is 600 psig. Therefore, the scram occurs well before valve closure begins.

J. Condenser Low Vacuum Scram

Loss of condenser vacuum occurs when the condenser can no longer handle the heat input. Loss of condenser vacuum initiates a closure of the turbine stop valves and turbine bypass valves which eliminates the heat input to the condenser. Closure of the turbine stop and bypass valves causes a pressure transient, neutron flux rise and an increase in surface heat flux. To prevent the cladding safety limit from being exceeded if this occurs, a reactor scram occurs on turbine stop valve closure in the Run mode. The turbine stop valve closure scram function alone is adequate to prevent the cladding safety limit from being exceeded in the event of a turbine trip transient with bypass closure. The condenser low vacuum scram is anticipatory to the stop valve closure scram and causes a scram before the stop valves are closed and thus the resulting transient is less severe. Scram occurs in the Run mode at 21-inch Hg vacuum, stop valve closure occurs at 20-inch Hg vacuum, and bypass closure at 7-inch Hg vacuum.

References

1. "Generic Reload Fuel Application," NEDE-24011-P-A.*

*Approved revision number at time reload analyses are performed.

2. "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors", General Electric Co. Licensing Topical Report NEDO 24154 Vols. I and II and NEDE-24154 Volume III as supplemented by letter dated September 5, 1980 from R.H. Buchholz (GE) to P.S. Check (NRC).

4.1 SURVEILLANCE REQUIREMENTS BASES

A. The minimum functional testing frequency used in this specification is based on a reliability analysis using the concepts developed in Reference 1. This concept was specifically adapted to the one-out-of-two taken twice logic of the reactor protection system. The analysis shows that the sensors are primarily responsible for the reliability of the reactor protection system. This analysis makes use of "unsafe failure" rate experience at conventional and nuclear power plants in a reliability model for the system. An "unsafe failure" is defined as one which negates channel operability and which, due to its nature, is revealed only when the channel is functionally tested or attempts to respond to a real signal. Failures such as blown fuses, ruptured bourdon tubes, faulted amplifiers, faulted cables, etc., which result in "upscale" or "downscale" readings on the reactor instrumentation are "safe" and will be easily recognized by the operators during operation because they are revealed by an alarm or a scram.

The channels listed in Tables 4.1-1 and 4.1-2 are divided into three groups respecting functional testing.

These are:

1. On-off sensors that provide a scram trip function (Group 1);

2. Analog devices coupled with bistable trips that provide a scram function (Group 2); and

3. Devices which serve a useful function only during some restricted mode of operation, such as Startup/Hot Standby, Refuel, or Shutdown, or for which the only practical test is one that can be performed at shutdown (Group 3).

The sensors that make up Group 1 are specifically selected from among the whole family of industrial on-off sensors that have earned an excellent reputation for reliable operation. Actual history on this class of sensors operating in nuclear power plants shows four failures in 472 sensor years, or a failure rate of 0.97×10^{-6}/hr. During design, a goal of 0.99999 probability of success (at the 50% confidence level) was adopted to assure that a balanced and adequate design is achieved. The probability of success is primarily a function of the sensor failure rate and the test interval. A 3-month test interval was planned for Group 1 sensors. This is in keeping with good operating practice and satisfies the design goal for the logic configuration utilized in the reactor protection system.

To satisfy the long-term objective of maintaining an adequate level of safety throughout the plant lifetime, a minimum goal of 0.9999 at the 95% confidence level is proposed. With the one-out-of-two taken twice logic, this requires that each sensor have an availability of 0.993 at the 95% confidence level. This level of availability may be maintained by adjusting the test interval as a function of the observed failure history (Reference 1). To facilitate the implementation of this technique, Figure 4.1-1 is provided to indicate an appropriate trend in test interval. The procedure is as follows:

1. Like sensors are pooled into one group for the purpose of data acquisition.

2. The factor M is the exposure hours and is equal to the number of sensors in a group, n, times the elapsed time T(M=nT).

3. The accumulated number of unsafe failures is plotted as an ordinate against M as an abscissa on Figure 4.1-1.

4. After a trend is established, the appropriate monthly test interval to satisfy the goal will be the test interval to the left of the plotted points.

5. A test interval of 1 month will be used initially until a trend is established.

The turbine control valve fast acting solenoid valve pressure switches directly measure the trip oil pressure that causes the turbine control valves to close in a rapid manner. The reactor scram setpoint was developed in accordance with NEDC 31336 "General Electric Instrument Setpoint Methodology" dated October, 1986. As part of the calculation, a calibration period is inputted to achieve a nominal trip point and an allowable setpoint (Technical Specification value). The nominal setpoint is procedurally controlled. Based on the calculation input, the calibration period is defined to be every Refueling Outage.

Group 2 devices utilize an analog sensor followed by an amplifier and a bistable trip circuit. The sensor and amplifier are active components, and a failure is almost always accompanied by an alarm and an indication of the source of trouble. In the event of failure, repair or substitution can start immediately. An "as-is" failure is one that "sticks" midscale and is not capable of going either up or down in response to an out-of-limits input. This type of failure for analog devices is a rare occurrence and is detectable by an operator who observes that one signal does not track the other three. For purposes of analysis, it is assumed that this rare failure will be detected within 2 hours.

The bistable trip circuit which is a part of the Group 2 devices can sustain unsafe failures which are revealed only on test. Therefore, it is necessary to test them periodically.

A study was conducted of the instrumentation channels included in the Group 2 devices to calculate their 'unsafe' failure rates. The analog devices (sensors and amplifiers) are predicted to have an unsafe failure rate of less than 20 X 10^{-6} failures/hour. The bistable trip circuits are predicted to have an unsafe failure rate of less than 2 X 10^{-6} failures/hour. Considering the 2-hour monitoring interval for the analog devices as assumed above and a weekly test interval for the bistable trip circuits, the design reliability goal of 0.99999 is attained with ample margin.

The bistable devices are monitored during plant operation to record their failure history and establish a test interval using the curve of Figure 4.1-1. There are numerous identical bistable devices used throughout the plant instrumentation system. Therefore, significant data on the failure rates for the bistable devices should be accumulated rapidly.

The frequency of calibration of the APRM flow biasing network has been established at each refueling outage. The flow biasing network is functionally tested at least once per month and, in addition, cross calibration checks of the flow input to the flow-biasing network can be made during the functional test by direct meter reading (IEEE 279 Standard for Nuclear Power Plant Protection Systems, Section 4.9, September 13, 1966). There are several instruments which must be calibrated, and it will take several days to perform the calibration of the entire network. While the calibration is being performed, a zero flow signal will be sent to half of the APRMs, resulting in a half scram and rod block condition. Thus, if the calibration were performed during operation, flux shaping would not be possible. Based on experience at other generating stations, drift of instrument such as those in the flow biasing network is not significant; therefore, to avoid spurious scrams, a calibration frequency of each refueling outage is established.

Reactor low water level instruments 2-263-57A, 2-263-57B, 2-263-58A, and 2-263-58B have been modified to be an analog trip system. The analog trip system consists of an analog sensor (transmitter) and a master/slave trip unit setup which ultimately drives a trip relay. The frequency of calibration and functional testing for instrument loops of the analog trip system, including reactor low water level, has been established in Licensing Topical Report NEDO-21617-A (December 1978). With the one-out-of-two-taken-twice logic, NEDO-21617-A states that each trip unit be subjected to a calibration/functional test frequency of one month. An adequate calibration/surveillance test interval for the transmitter is once per operating cycle.

Group 3 devices are active only during a given portion of the operational cycle. For example, the IRM is active during startup and inactive during full-power operation. Thus, the only test that is meaningful is the one performed just prior to shutdown or startup, i.e., the tests that are performed just prior to use of the instrument.

Calibration frequency of the instrument channel is divided into two groups. These are as follows:

1. Passive type indicating devices that can be compared with like units on a continuous basis, and

2. Vacuum tube or semiconductor devices and detectors that drift or lose sensitivity.

Experience with passive type instruments in Commonwealth Edison generating stations and substations indicate that the specified calibrations are adequate. For those devices which employ amplifiers, etc., drift specifications call for drift to be less than 0.4%/month, i.e., in the period of a month a drift of 0.4% would occur, thus providing for adequate margin.

The sensitivity of LPRM detectors decreases with exposure to neutron flux at a slow and approximately constant rate. Changes in power distribution and electronic drift also require compensation. This compensation is accomplished by calibrating the APRM system every 7 days using heat balance data and by calibrating individual LPRMs at least every 1000 equivalent full-power hours using TIP traverse data. Calibration on this frequency assures plant operation at or below thermal limits.

A comparison of Tables 4.1-1 and 4.1-2 indicates that some instrument channels have not been included in the latter table. These are mode switch in shutdown, manual scram, high water level in scram discharge volume, main steamline isolation valve closure, and turbine stop valve closure. All of the devices or sensors associated with these scram functions are simple on-off switches, hence calibration is not applicable, i.e., the switch is either on or off. Further, these switches are mounted solidly to the device and have a very low probability of moving; e.g., the thermal switches in the scram discharge volume tank. Based on the above, no calibration is required for these instrument channels.

B. The MFLPD shall be checked once per day to determine if the APRM scram requires adjustment. This may normally be done by checking the LPRM readings, TIP traces, or process computer calculations. Only a small number of control rods are moved daily, thus the peaking factors are not expected to change significantly and a daily check of the MFLPD is adequate.

References

1. I. M. Jacobs, "Reliability of Engineered Safety Features as a Function of Testing Frequency," Nuclear Safety, Vol. 9, No. 4, pp. 310-312, July-August, 1968.

2. Licensing Topical Report NEDO-21617-A (December 1978).

3. NEDC - 31336 "General Electric Instrument Setpoint Methodology" dated October, 1986.

TABLE 3.1-3

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENTS RUN MODE

Minimum Number of Operable or Tripped Instrument Channels per Trip System[1]	Trip Function	Trip Level Setting	Action[2]
1	Mode switch in shutdown		A
1	Manual scram		A
	APRM[3]		
2	High Flux (flow biased)	Specification 2.1.A.1	A or B
2	Inoperative		A or B
2	Downscale [11]	≥ 3/125 of full scale	A or B
2	High-reactor pressure	≤ 1060 psig	A
2	High drywell pressure	≤ 2.5 psig	A
2	Reactor low water level	≥ 8 inches[8]	A
2 (per bank)	High-water level in scram discharge volume	≤ 40 gallons per bank	A
2	Turbine condenser low vacuum	≥ 21 inches Hg vacuum	A or C
2	Main Steamline high radiation [12]	≤ 15 X normal full power background (without hydrogen addition)	A or C
4	Main steamline isolation valve closure [6]	≤ 10% valve closure	A or C
2	Turbine control valve fast closure, valve trip system oil pressure low [9]	≥ 460 psig [10]	A or C
2	Turbine stop valve closure [9]	≤ 10% valve closure	A or C
2	Turbine EHC control fluid low pressure [9]	≥ 900 psig	A or C

TABLE 3.1-4

NOTES FOR TABLES 3.1-1, 3.1-2, AND 3.1-3

[1] There shall be two operable trip systems or one operable and one tripped system for each function.

[2] If the first column cannot be met for one of the trip systems, that trip system shall be tripped. If the first column cannot be met for both trip systems, the appropriate actions listed below shall be taken:

A. Initiate insertion of operable rods and complete insertion of all operable rods within 4 hours.

B. Reduce power level to IRM range and place mode switch in the Startup/Hot Standby position within 8 hours.

C. Reduce turbine load and close main steamline isolation valves within 8 hours.

[3] An APRM will be considered inoperable if there are fewer than 2 LPRM inputs per level or there are less than 50% of the normal complement of LPRM's to an APRM.

[4] Permissible to bypass, with control rod block for reactor protection system reset in refuel and shutdown positions of the reactor mode switch.

[5] Not required to be operable when primary containment integrity is not required.

[6] The design permits closure of any one line without a scram being initiated.

[7] Automatically bypassed when reactor pressure is < 1060 psig.

[8] The +8-inch trip point is the water level as measured by the instrumentation outside the shroud. The water level inside the shroud will decrease as power is increased to 100% in comparison to the level outside the shroud to a maximum of 7 inches. This is due to the pressure drop across the steam dryer. Therefore, at 100% power, an indication of +8-inch water level will actually be +1 inch inside the shroud. 1 inch on the water level instrumentation is $\geq$ 504" above vessel zero. (See Bases 3.2).

[9] Permissible to bypass when first stage turbine pressure is less than that which corresponds to 45% rated steam flow. (< 400 psi)

[10] Trip is indicative of turbine control valve fast closure (due to low EHC fluid pressure) as a result of fast acting valve actuation.

TABLE 3.1-4

NOTES FOR TABLES 3.1-1, 3.1-2, AND 3.1-3 (Continued)

[11] The APRM downscale trip function is automatically bypassed when the IRM instrumentation is operable and not high.

[12] Channel shared by the reactor protection and containment isolation system.

TABLE 4.1-2

SCRAM INSTRUMENT CALIBRATION
MINIMUM CALIBRATION FREQUENCIES FOR REACTOR PROTECTION INSTRUMENT CHANNELS

Instrument Channel	Group[1]	Calibration Standard[5]	Minimum Frequency[2]
High flux IRM	C	Comparison to APRM after heat balance	Every controlled shutdown[4]
High flux APRM			
Output signal	B	Heat balance	Once every 7 days
Flow bias	B	Standard pressure and voltage source	Refueling outage
LPRM	B[6]	Using TIP system	Every 1000 equivalen full power hours
High reactor pressure	A	Standard pressure source	Every 3 months
High drywell pressure	A	Standard pressure source	Every 3 months
Reactor low water level	B	Water level	[7]
Turbine condenser low vacuum	A	Standard vacuum source	Every 3 months
Main steamline high radiation	B	Appropriate radiation source [3]	Refueling outage
Turbine EHC control fluid low pressure	A	Pressure source	Every 3 months
Turbine control valve fast closure	A	Pressure source	Refueling outage
Highwater level in scram discharge volume (dp only)	A	Water level	Refueling outage

Notes:

[1] A description of the three groups is included in the bases of this specification.

[2] Calibration tests are not required when the systems are not required to be operable or are tripped. If tests are missed, they shall be performed prior to returning the systems to an operable status.

[3] A current source provides an instrument channel alignment every 3 months.

TABLE 4.1-2

SCRAM INSTRUMENT CALIBRATION
MINIMUM CALIBRATION FREQUENCIES FOR REACTOR PROTECTION INSTRUMENT CHANNELS

Notes: (Continued)

[4] Maximum calibration frequency need not exceed once per week.

[5] Response time is not part of the routine instrument check and calibration but will be checked every refueling outage.

[6] Does not provide scram function.

[7] Trip units are calibrated monthly concurrently with functional testing. Transmitters are calibrated once per operating cycle.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 125 TO FACILITY OPERATING LICENSE NO. DPR-30

COMMONWEALTH EDISON COMPANY

AND

IOWA-ILLINOIS GAS AND ELECTRIC COMPANY

QUAD CITIES NUCLEAR POWER STATION, UNIT 2

DOCKET NO. 50-265

1.0 INTRODUCTION

By letter dated April 18, 1991, Commonwealth Edison Company (the licensee) proposed changes to the Technical Specifications (TS) for Quad Cities Nuclear Power Station, Unit 2. The proposed changes reflect a modification to the fast acting solenoid valves which initiate rapid closure of the turbine control valves. The new design uses a pressure switch, rather than a limit switch, to initiate a reactor scram. A similar amendment was issued for Unit 1 on February 21, 1991.

2.0 DISCUSSION AND EVALUATION

The staff evaluated the licensee's submittal in four distinct areas:

a. Pressure switch vs. limit switch function - the new design must not change the original intent of the function.

b. Setpoint calculation - the calculation has to be developed using well developed standards.

c. Surveillance interval - the calibration interval must be consistent with the design and the setpoint calculation.

d. TS changes - the proposed TSs must reflect the new design.

The objective of the turbine control fast acting solenoid valves is to protect the turbine from overspeed when the load is suddenly removed. The logic to determine the load reduction is the load control unit of the Electro-Hydraulic Control System. The load unbalance signal activates relays which send a signal to the turbine fast acting solenoid valves. Actuation of the fast acting solenoid valves inputs to the Reactor Protection System to provide a reactor scram. The objective of the scram

4pp

is to anticipate the rapid increase in the pressure and neutron flux which may result from the fast closure of the turbine control valves and subsequent failure of the turbine steam bypass valves.

The existing design uses a limit switch off the fast acting solenoid valves to initiate a reactor scram. The limit switch is a simple on-off status indicator that changes status depending on the position of the fast acting solenoid valves. There is no periodic calibration of the on-off devices and there is no setpoint calculations associated with the limit switch. The existing fast acting solenoid valves have failed several times during surveillance testing. Because of these failures, the licensee has decided to replace the valves with new valves as recommended by the NSSS vendor, General Electric. The new fast acting solenoid valves are manufactured by Parker-Hannifan and have been used with good results since 1976. The function of the new solenoid valves remains the same. However, the reactor scram is now initiated by a pressure switch instead of a limit switch.

The new design does introduce a new failure mode due to the tubing which connects the pressure switch to the solenoid valve. Rupture of the tubing would initiate a reactor scram. However, this is a conservative action. The industry use of the pressure switch as input to the Reactor Protection System has proven more reliable than the existing limit switch. Therefore, the function of the pressure switch to replace the existing limit switch is acceptable.

The new fast acting solenoid valves require a determination of the pressure setpoint. General Electric, in its generic instrument setpoint methodology (NEDC-31336, October 1986), addressed the Turbine Control Valve Fast Closure. The licensee has followed the General Electric methodology. The staff is presently reviewing the General Electric generic setpoint methodolcgy, but has not completed the review. Consequently, the Quad Cities setpoint calculation has been reviewed on an individual basis. The ongoing General Electric methodology evaluation has been found acceptable for this particular setpoint calculation. Accordingly, the Quad Cities calculation was reviewed for consistency with the General Electric generic setpoint calculation.

The pressure switches directly measure the trip oil pressure that causes the turbine control valves to close in a rapid manner. This oil pressure is normally about 1500 to 1600 psig, and the control valve does not start to close until the pressure drops to 400 psig. It is considered possible in normal operations for the pressure to drop to 740 psig due to transients. Therefore, the analytical limit is 400 psig, and the operational limit is 740 psig.

The instrument accuracy of the pressure switch is two percent of full scale. Accuracy is conservatively estimated to be one percent of full range. Full scale is 3000 psig. The instrument drift for a six-month interval is equal to the instrument accuracy. Drift is assumed to be random and calculated to be ±104 psig for an 18-month refueling outage. The licensee reported the drift that was actually experienced by the same pressure switches at other plants. The data indicate that the drift assumed by the licensee is conservative.

Using the above data and the methodology of NEDC-31336, General Electric performed the pressure setpoint calculation. The setpoint calculation determined the allowable value or TS value of 460 psig and the nominal trip setpoint of 590 psig. The staff concludes that the setpoint calculation is consistent with the General Electric setpoint methodology and, therefore, is acceptable.

The proposed fast acting solenoid valves are designed for the pressure switch to be actuated within 30 milliseconds of the time the control valves begin to close. This time is consistent with the design values used in the reload licensing calculations to analyze the load reject without bypass valve transient. Verification of the 30 milliseconds actuation will be conducted during post-modification testing. Therefore, this modification does not involve a reduction in the margin of safety as previously determined.

The proposed calibration frequency is every refueling outage. This proposed frequency is consistent with the guidance in NUREG-0123, "General Electric Standard Technical Specifications," Revision 4. This interval is also consistent with TS for BWR plants licensed in the 1980s. General Electric used the 18-month interval in the pressure setpoint calculation. This frequency is used by General Electric in its generic setpoint methodology (NEDC-31336). Therefore, the staff concludes that the surveillance interval for the turbine control valve fast closure is acceptable.

The proposed TS change revises Table 4.1-2 to require that the fast acting solenoid valves pressure switch be calibrated every refueling outage. Page 3.1/4.1-10 of the TSs is revised to delete the description of the turbine control valve fast closure scram device as a simple on-off switch. Table 3.1-3 and Section 2.1.F are revised to accurately define the trip level setting of the turbine control valve fast closure scram to greater than 460 psig Electro-Hydraulic Control oil pressure. In addition, the appropriate sections to the Bases are provided to reflect the new design of the fast acting solenoid valves. Therefore, the staff concludes that the proposed TS changes reflect the new design of the turbine control fast acting solenoid valves.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Illinois State official was notified of the proposed issuance of the amendments. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (56 FR 24206). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

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