Docket No. 50-334

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Mr. C. N. Dunn, Vice President
Operations Division
Duquesne Light Company
435 Sixth Avenue
Pittsburgh, Pennsylvania 15219

Dear Mr. Dunn:

The Commission has issued the enclosed Amendment No. 39 to Facility Operating License No. DPR-66 for the Beaver Valley Nuclear Power Station, Unit No. 1. The amendment consists of changes to the Technical Specifications in response to your application transmitted by letter dated September 17, 1980.

The amendment adds license conditions 2.C.(7), 2.C.(8) and 2.C.(9) and revises the Technical Specifications to incorporate TMI-2 Lessons Learned Category "A" items. This amendment satisfies all Category "A" requirements for Beaver Valley Nuclear Power Station, Unit No. 1.

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

Sincerely,

Original signed by:
S. A. Varga

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Enclosures:
1. Amendment No. 39 to DPR-66
2. Safety Evaluation
3. Notice of Issuance

cc w/encl:
See next page

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Docket

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

February 11, 1981

Docket No. 50-334

Mr. C. N. Dunn, Vice President
Operations Division
Duquesne Light Company
435 Sixth Avenue
Pittsburgh, Pennsylvania 15219

Dear Mr. Dunn:

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Sincerely,

Steven A. Varga

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Operating Reactors Branch #1
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See next page

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

DUQUESNE LIGHT COMPANY

OHIO EDISON COMPANY

PENNSYLVANIA POWER COMPANY

DOCKET NO. 50-334

BEAVER VALLEY POWER STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 39
License No. DPR-66

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

   A. The application for amendment by Duquesne Light Company, Ohio Edison Company, and Pennsylvania Power Company (the licensees) dated September 17, 1980, 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. DPR-66 is hereby amended by:

   a. Renumbering paragraphs 2.C.(4), 2.C.(5), 2.C.(8) and 2.C.(9) as 2.C.(3), 2.C.(4), 2.C.(5) and 2.C.(6), respectively.

   b. Adding the following paragraphs 2.C.(7), 2.C.(8) and 2.C.(9):

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(7) Systems Integrity

Duquesne Light Company shall implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. This program shall include the following:

1. Provisions establishing preventive maintenance and periodic visual inspection requirements, and

2. Integrated leak test requirements for each system at a frequency not to exceed refueling cycle intervals.

(8) Iodine Monitoring

Duquesne Light Company shall implement a program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

1. Training of personnel,

2. Procedures for monitoring, and

3. Provisions for maintenance of sampling and analysis equipment.

(9) Backup Method for Determining Subcooling Margin

Duquense Light Company shall implement a program which will ensure the capability to accurately monitor the Reactor Coolant System subcooling margin. This program shall include the following:

1. Training of personnel, and

2. Procedures for monitoring.

3. Additionally, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-66 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 39 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 11, 1981

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 39 TO FACILITY OPERATING LICENSE NO. DPR-66

DOCKET NO. 50-334

Revise Appendix A as follows:

Remove Pages	Insert Pages
iv	iv
v	v
-	3/4 3-19a
-	3/4 3-24a
-	3/4 3-27a
-	3/4 3-31a
-	3/4 3-50
-	3/4 3-51
-	3/4 3-52
3/4 4-7	3/4 4-7
-	3/4 4-31
B 3/4 3-3	B 3/4 3-3
B 3/4 4-2	B 3/4 4-2
B 3/4 4-10	B 3/4 4-10
6-4	6-4
6-5	6-5

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

LIMITING CONDITIONS FOR OPER' ION AND SURVEILLANCE REQUIREMENTS

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
7. AUXILIARY FEEDWATER					
a. Steam. Gen. Water Level-Low-Low					
i. Start Turbine Driven Pump	3/stm. gen.	2/stm. gen. any stm. gen.	2/stm. gen	1, 2, 3	14
ii. Start Motor Driven Pumps	3/stm. gen.	2/stm. gen. any 2 stm. gen.	2/stm. gen.	1, 2, 3	14
b. Undervoltage-RCP Start Turbine-Driven Pump	(3)-1/bus	2	2	1	14
c. S. I. Start Motor-Driven Pumps	See 1 above (all S.I. initiating functions and requirements)				
d. Emergency Bus Undervoltage Start Motor Driven Pumps	1/bus	1	1	1, 2, 3	18
e. Trip of Main Feedwater Pumps Start Motor-Driven Pumps	1/pump	1	1	1, 2, 3	18

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
7. AUXILIARY FEEDWATER		
a. Steam Generator Water Level-low-low	≥ 12% of narrow range instrument span each steam generator	≥ 11% of narrow range instrument span each steam generator
b. Undervoltage - RCP	≥ 2750 volts RCP bus voltage	≥ 2725 volts RCP bus voltage
c. S.I.	See 1 above (all SI Setpoints)	
d. Emergency Bus Undervoltage	≤ 3350 volts	≤ 3325 volts
e. Trip of Main Feedwater Pumps	N/A	N/A

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
11. Steam Generator Water Level-Low-low	
a. Motor-driven Auxiliary Feedwater Pumps**	60.0
b. Turbine-driven Auxiliary Feedwater Pumps***	60.0
12. Undervoltage RCP	
a. Turbine-driven Auxiliary Feedwater Pumps	60.0
13. Emergency Bus Undervoltage	
a. Motor-driven Auxiliary Feedwater Pumps	60.0
14. Trip of Main Feedwater Pumps	
a. Motor-driven Auxiliary Feedwater Pumps	60.0
Note: Response time for Motor-driven Auxiliary Feedwater Pumps on all S.I. signal starts	60.0

***on 2/3 any Steam Generator
**on 2/3 in 2/3 Steam Generators

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
7. AUXILIARY FEEDWATER				
a. Steam Generator Water Level-Low-Low	S	R	M	1, 2, 3
b. Undervoltage - RCP	S	R	M	1, 2
c. S.I.	See 1 above (all SI surveillance requirements)			
d. Emergency Bus Undervoltage	N/A	R	R	1, 2, 3
e. Trip of Main Feedwater Pumps	N/A	N/A	R	1, 2, 3

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.8 The accident monitoring instrumentation channels shown in Table 3.3.11 shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With the number of OPERABLE accident monitoring instrumentation channels less than the Total Number of Channels shown in Table 3.3.11, either restore the inoperable channel(s) to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours except for the PORV(s) which may be isolated in accordance with Specification 3.4.11.a.

b. With the number of OPERABLE accident monitoring instrumentation channels less than the MINIMUM CHANNELS OPERABLE requirements of Table 3.3.11, either restore the inoperable channel(s) to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.8 Each accident monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-7.

TABLE 3.3-11

ACCIDENT MONITORING INSTRUMENTATION

	INSTRUMENT	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE
1.	Pressurizer Water Level	(3)	(2)
2.	Auxiliary Feedwater Flow Rate	(1) per steam gen.	(1) per steam gen.
3.	Reactor Coolant System Subcooling Margin Monitor	(1)	(0)
4.	PORV Accoustical Detector Position Indicator	2/valve*	1/valve
5.	PORV Limit Switch Position Indicator	1/valve	0/valve
6.	PORV Block Valve Limit Switch Position Indicator	1/valve	0/valve
7.	Safety Valve Accoustical Detector Position Indicator	2/valve*	1/valve
8.	Safety Valve Temperature Detector Position Indicator	1/valve	0/valve

* One Detector Active, Second Detector Passive

TABLE 4.3-7

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

	INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1.	Pressurizer Water Level	M	R
2.	Auxiliary Feedwater Flow Rate	S/U(1)	R
3.	Reactor Coolant System Subcooling Margin Monitor	M	R
4.	PORV Accoustical Detector Position Indicator	M	R
5.	PORV Limit Switch Position Indicator	M	R
6.	PORV Block Valve Limit Switch Position Indicator	M	R
7.	Safety Valve Accoustical Detector Position Indicator	M	R
8.	Safety Valve Temperature Detector Position Indicator	M	R

(1) Channel check to be performed in conjunction with Surveillance Requirement 4.7.1.2.a.9 following an extended plant outage.

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.4 The pressurizer shall be OPERABLE with at least (150) **kw** of pressurizer heaters and with a steam bubble.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With the pressurizer inoperable due to less than 150 kw of heaters supplied by an emergency bus, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 12 hours. With the pressurizer otherwise inoperable, be in at least HOT STANDBY with the reactor trip breakers open within 6 hours and in the HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.4.1 The emergency power supply for the pressurizer heaters shall be demonstrated OPERABLE at least once per 18 months by energizing the heaters supplied by the emergency bus.

3/4 4.11 RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.11 (Two) power operated relief valves (PORVs) and their associated block valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With less than 2 PORV(s) operable, within 1 hour either restore two PORV(s) to OPERABLE status or close the associated block valve(s) and remove power from the block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one or more block valve(s) inoperable, within 1 hour either restore the block valve(s) to OPERABLE status or close the block valves(s) and remove power from the block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.11.1 Each PORV shall be demonstrated OPERABLE:

a. At least once per 31 days by performance of a CHANNEL CHECK of the position indication, excluding valve operation and

b. At least once per 18 months by performance of a CHANNEL CALIBRATION.

4.4.11.2 Each block valve shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel.

4.4.11.3 The emergency power supply for the PORVs and block valves shall be demonstrated OPERABLE at least once per 18 months by operating the valves through a complete cycle of full travel.

3/4.3.3.5 REMOTE SHUTDOWN INSTRUMENTATION

The OPERABILITY of the remote shutdown instrumentation ensures that sufficient capability is available to permit shutdown and maintenance of HOT STANDBY of the facility from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criteria 19 of 10 CFR 50.

3/4.3.3.6 FIRE DETECTION INSTRUMENTATION

OPERABILITY of the fire detection instrumentation ensures that adequate warning capability is available for the prompt detection of fires. This capability is required in order to detect and locate fires in their early stages. Prompt detection of fires will reduce the potential for damage to safety related equipment and is an integral element in the overall facility fire protection program.

In the event that a portion of the fire detection instrumentation is inoperable, the establishment of frequent fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABILITY.

3/4.3.3.8 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables during and following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Plants to Assess Plant Conditions During and Following an Accident," December 1975 and NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations."

relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating RHR loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization.

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its safety limit of 2735 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from a complete loss of load assuming no reactor trip until the first Reactor Protective System trip set point is reached (i.e., no credit is taken for a direct reator trip on the loss of load) and also assuming no operation of the power operated relief valves or steam dump valves.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Subsection IWV-3510 of Section XI of the ASME Boiler and Pressure Code, dated July 1974.

3/4.4.4 PRESSURIZER

The requirement that (150)kw of pressurizer heaters and their associated controls be capable of being supplied electrical power from an emergency bus provides assurance that these heaters can be energized during a loss of offsite power condition to maintain natural circulation at HOT STANDBY.

3/4.4.5 STEAM GENERATORS

One OPERABLE steam generator in a non-isolated reactor coolant loop provides sufficient heat removal capability to remove decay heat after a reactor shutdown. The requirement for two OPERABLE steam generators, combined with other requirements of the Limiting Conditions for Operation ensures adequate decay heat removal capabilities for RCS temperatures greater than 350°F if one steam generator becomes inoperable due to single failure considerations. Below 350°F, decay heat is removed by the RHR system.

vessel inside radius are essentially identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of the reactor vessel. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule is different from the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

The pressure-temperature limit lines shown on Figure 3.4-2 for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR 50 for reactor criticality and for inservice leak and hydrostatic testing.

The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided in Table 4.4-3 to assure compliance with the requirements of Appendix H to 10 CFR Part 50.

The limitations imposed on the pressurizer heatup and cooldown rates and spray water temperature differential are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements.

3/4.4.10 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for ASME Code Class 1, 2 and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR Part 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 50.55a (g)(6)(i).

3/4.4.11 RELIEF VALVES

The relief valves have remotely operated block valves to provide a positive shutoff capability should a relief valve become inoperable. The electrical power for both the relief valves and the block valves is capable of being supplied from an emergency power source to ensure the ability to seal this possible RCS leakage path.

TABLE 6.2-1

MINIMUM SHIFT CREW COMPOSITION#

SINGLE UNIT FACILITY

LICENSE CATEGORY QUALIFICATIONS	APPLICABLE MODES 1, 2, 3 and 4	5 and 6
SRO*	2	1**
RO	2	1
Non-Licensed Auxiliary Operator	2	1
Shift Technical Advisor	1	None Required

* Includes the licensed Senior Reactor Operator serving as the Shift Supervisor.

** Does not include the licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling, supervising CORE OPERATIONS.

Shift crew composition may be one less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Table 6.2-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent.

6.3 FACILITY STAFF QUALIFICATIONS

6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for the Radiation Control Supervisor who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975, and the Shift Technical Advisor who shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design and response analysis of the plant for transients and accidents.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the facility staff shall be maintained under the direction of the Training Supervisor and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and Appendix "A" of 10 CFR Part 55.

6.4.2 A Training program for the Emergency Squad shall be maintained under the direction of the Training Supervisor and shall meet or exceed the requirements of Section 27 of the NEPA Code-1976.

6.5 REVIEW AND AUDIT

6.5.1 ONSITE SAFETY COMMITTEE (OSC)

FUNCTION

6.5.1.1 The OSC shall function to advise the Plant Superintendent on all matters related to nuclear safety.

COMPOSITION

6.5.1.2 The OSC shall be composed of the:

Chairman:	Chief Engineer
Member:	Operations Supervisor
Member:	Radiation Control Supervisor
Member:	Maintenance Supervisor
Member:	Nuclear Engineering & Refueling Supervisor
Member:	Results Coordinator
Member:	Training Supervisor
Member:	Office Manager Nuclear (Security Officer)
Member:	Senior Engineer - Emergency Planning and Fire Protection
Member:	Technical Advisory Engineer

ALTERNATES

6.5.1.3 All alternate members shall be appointed in writing by the OSC Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in OSC activities at any one time.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 39 TO FACILITY OPERATING LICENSE NO. DPR-66

DUQUESNE LIGHT COMPANY

OHIO EDISON COMPANY

PENNSYLVANIA POWER COMPANY

BEAVER VALLEY POWER STATION, UNIT NO. 1

DOCKET NO. 50-334

INTRODUCTION

By letter dated September 17, 1980, Duquesne Light Company proposed changes to the Technical Specifications (TSs) appended to Facility Operating License No. DPR-66 for the Beaver Valley Power Station, Unit No. 1. The changes involve the incorporation of certain of the TMI-2 Lessons Learned Category "A" requirements. The licensee's request is in direct response to the NRC staff's letter dated July 2, 1980.

BACKGROUND INFORMATION

By our letter dated September 13, 1979, we issued to all operating nuclear power plants requirements established as a result of our review of the TMI-2 accident. Certain of these requirements, designated Lessons Learned Category "A" requirements, were to have been completed by the licensee prior to any operation subsequent to January 1, 1980. Our evaluation of the licensee's compliance with these Category "A" items was attached to our letter to Duquesne Light Company dated October 9, 1980.

In order to provide reasonable assurance that operating reactor facilities are maintained within the limits determined acceptable following the implementation of the TMI-2 Lessons Learned Category "A" items, we requested that licensees amend their TS to incorporate additional Limiting Conditions of Operation and Surveillance Requirements, as appropriate. This request was transmitted to all licensees on July 2, 1980. Included therein were model specifications that we had determined to be acceptable. The licensee's application is in direct response to our request. Each of the issues identified by the NRC staff and the licensee's response is discussed in the Evaluation below.

EVALUATION

2.1.1 Emergency Power Supply Requirements

The pressurizer water level indicators, pressurizer relief and block valves, and pressurizer heaters are important in a post-accident situation. Adequate emergency power supplies add assurance of post-accident functioning of these components. The licensee has the requisite emergency power supplied. The licensee has proposed adequate TSs which provide for a 31-day channel check and 18-month channel calibration and actions in the event of component inoperability. We have reviewed these proposed TSs and find that the emergency power supplies are reasonably ensured for post-accident functioning of the subject components and are thus acceptable.

2.1.3.a Direct Indication of Valve Position

The licensee has provided a direct indication of power-operated relief valve (PORV) and safety valve position in the control room and direct indication of flow downsteam of the PORV and safety valves in the control room. These indications are a diagnostic aid for the plant operator and provide no automatic action. The licensee has provided TSs with a 31-day channel check and an 18-month channel calibration requirement; thus, the TSs are acceptable and they meet our July 2, 1980 model TS criteria.

2.1.3.b Instrumentation for Inadequate Core Cooling

The licensee has installed an instrument system to detect the effects of low reactor coolant level and inadequate core cooling. This instrument, a sub-cooling meter, receives and processes data from existing plant instrumentation. We previously reviewed this system in our Safety Evaluation dated October 9, 1980. The licensee submitted TSs with a 31-day channel check and an 18-month channel calibration requirement and actions to be taken in the event of component inoperability. We conclude that TSs are acceptable as they meet our July 2, 1980 model TS criteria.

2.1.4 Diverse Containment Isolation

The licensee has modified the containment isolation system so that diverse parameters will be sensed to ensure automatic isolation of non-essential systems under postulated accident conditions. We have reviewed this system in our Lessons Learned Category "A" Safety Evaluation dated October 9, 1980. The modification is such that it does not result in the automatic loss of containment isolation after the containment isolation signal is reset. Reopening of containment isolation would require deliberate operator action. The TSs submitted by the licensee list each affected containment isolation valve and provide for the appropriate surveillance and actions in the event of component inoperability; therefore, we conclude that the TSs are acceptable.

2.1.7.a Auto Initiation of Auxiliary Feedwater Systems

The licensee has provided for the automatic initiation of auxiliary feedwater flow on loss of normal feedwater flow. The auto-initiation signals used by the licensee are steam generator level and safety injection. We have previously reviewed the design and installation of this system as part of our Lessons Learned Category "A" program. The circuits are designed to be testable and the design retains the capability of manual actuation from the control room even in the event of failure of the auto-initiating circuitry. The TSs submitted by the licensee list the appropriate components, describe the tests and provide for proper test frequency. The TSs contain appropriate actions in the event of component inoperability; therefore, we conclude that the TSs are acceptable.

2.1.7.b Auxiliary Feedwater Flow Indication

The licensee has installed auxiliary feedwater flow indication that meets our testability and vital power requirements. We reviewed this system in our Safety Evaluation dated October 9, 1980. The licensee has proposed a TS with 31-day channel check and 18-month channel calibration requirements. We find this TS acceptable as it meets the criteria of our July 2, 1980 model TS criteria.

2.2.1.b Shift Technical Advisor (STA)

Our request indicated that the TSs related to minimum shift manning should be revised to reflect the augmentation of an STA. The licensee's application would add one STA to each shift to perform the function of accident assessment. The individual performing this function will have at least a bachelor's degree or equivalent in a scientific or engineering discipline with special training in plant design, and response and analysis of the plant for transients and accidents. Part of the STA duties are related to operating experience review function. Based on our review, we find the licensee's submittal to satisfy our requirements and is acceptable.

EVALUATION TO SUPPORT LICENSE CONDITIONS

2.1.4 Integrity of Systems Outside Containment

Our letter dated July 2, 1980, indicated that the license should be amended by adding a license condition related to a Systems Integrity Measurements Program. Such a condition would require the licensee to effect an appropriate program to eliminate or prevent the release of significant amounts of radioactivity to the environment via leakage from engineered safety systems and auxiliary systems, which are located outside reactor containment. By letter dated September 17, 1980, the licensee agreed to adopt such a license condition; accordingly we have included this condition in the license.

2.1.8.c Iodine Monitoring

Our letter dated July 2, 1980, indicated that the license should be amended by adding a license condition related to iodine monitoring. Such a condition would require the licensee to effect a program which would ensure the capability to determine the airborne iodine concentration in areas requiring personnel access under accident conditions. By letter dated September 17, 1980, the licensee agreed to adopt such a license condition; accordingly, we have included this condition in the license.

2.1.3.b Backup Method for Determining Subcooling Margin

Our letter of July 2, 1980, indicated that the license should be amended by adding a license condition related to the determination of subcooling margin; this is a precursor to warn of inadequate core cooling in the event of an accident. Such a condition would require the training of personnel and the generation of procedures to accurately monitor the reactor coolant system subcooling margin. By letter dated September 17, 1980, the licensee agreed to adopt such a license condition; accordingly, we have included this condition in the license.

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 the amendment.

CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) because that 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.

Date: February 11, 1981

7590-01

UNITES STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-334

DUQUESNE LIGHT COMPANY

OHIO EDISON COMPANY

PENNSYLVANIA POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U.S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 39 to Facility Operating License No. DPR-66 issued to Duquesne Light Company, Ohio Edison Company, and Pennsylvania Power Company (the licensees), which revised Technical Specifications for operation of the Beaver Valley Power Station, Unit No. 1 (the facility) located in Beaver County, Pennsylvania. The amendment is effective as of the date of issuance.

The amendment revises the Technical Specifications to incorporate TMI-2 Lessons Learned Category "A" items.

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 this 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 issuance of this amendment.

For further details with respect to this action, see (1) the application for amendment dated September 17, 1980, (2) Amendment No. 39 to License No. DPR-66 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, NW., Washington, D.C. and at the B. F. Jones Memorial Library, 663 Franklin Avenue, Aliquippa, Pennsylvania 15001. 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 11th day of February, 1981.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing