September 28, 1989

Docket No. 50-334

Mr. J. D. Sieber, Vice President
Nuclear Group
Duquesne Light Company
Post Office Box 4
Shippingport, Pennsylvania 15077

Dear Mr. Sieber:

SUBJECT: BEAVER VALLEY UNIT 1 - ISSUANCE OF AMENDMENT (TAC NO. 73123)

The Commission has issued the enclosed Amendment No. 144 to Facility Operating License No. DPR-66 for the Beaver Valley Power Station, Unit No. 1, in response to your application dated May 9, 1989, and supplement dated July 21, 1989.

The amendment revises the Technical Specifications and associated basis pages to permit use of upgraded Westinghouse fuel design in fuel cycle 8 and beyond. The upgraded features include the Vantage 5H design features, reconstitutable top nozzles, debris filter bottom nozzles, snag resistant grids and standardized fuel pellets.

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

Sincerely,

original signed by

Peter S. Tam, Senior Project Manager
Project Directorate I-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 144 to DPR-66
2. Safety Evaluation

cc w/enclosures:
See next page

[BV AMEND 73123]

LA:PDI-4	PM:PDI-4	D:PDI-4	OGC
SNorris	PTam:bld PST	JStolz PSTam for	BMB
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DATED: September 28, 1989

AMENDMENT NO. 144 TO FACILITY OPERATING LICENSE NO. DPR-66 - BEAVER VALLEY, UNIT 1

Docket File
NRC & Local PDRs
PDI-4 Reading
SVarga, 14/E/4
BBoger, 14/A/2
JStolz
SNorris
PTam
OGC (for information only)
DHagan, 3302 MNBB
EJordan, 3302 MNBB
BGrimes, 9/A/2
TMeek (4)
Wanda Jones
HCalvo 11/F/23
ACRS (10)
GPA/PA
OC/LFMB
SRhow

cc: Plant Service list

DF01
1/1

Mr. J. Sieber
Duquesne Light Company

Beaver Valley Power Station
Units 1 & 2

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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. 144
License No. DPR-66

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

   A. The application for amendment by Duquesne Light Company, et al. (the licensee) dated May 9, 1989 and supplement dated July 21, 1989, 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.

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2. Accordingly, 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 Appendix A, as revised through Amendment No. 144, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective on issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Peter S. Tam for

John F. Stolz, Director
Project Directorate I-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 28, 1989

ATTACHMENT TO LICENSE AMENDMENT NO. 144

FACILITY OPERATING LICENSE NO. DPR-66

DOCKET NO. 50-334

Replace the following pages of Appendix A, Technical Specifications, with the enclosed pages as indicated. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change.

Remove	Insert
B 2-1	B 2-1
B 2-2	B 2-2
B 2-4	B 2-4
B 2-6	B 2-6
3/4 1-22	3/4 1-22
3/4 2-5	3/4 2-5
3/4 2-7	3/4 2-7
3/4 2-8	3/4 2-8
B 3/4 2-1	B 3/4 2-1
B 3/4 2-4	B 3/4 2-4
B 3/4 2-5	B 3/4 2-5
B 3/4 2-6	B 3/4 2-6
B 3/4 4-1	B 3/4 4-1
B 3/4 9-4	B 3/4 9-4

2.1.1 REACTOR CORE

The restrictions of this safety limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB through the WRB-1 correlation. The WRB-1 DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

The DNB design basis is as follows: there must be at least a 95 percent probability that the minimum DNBR of the limiting fuel rod during Condition I and II events is greater than or equal to the DNBR limit of the DNB correlation being used (the WRB-1 correlation in this application). The correlation DNBR limit is established based on the entire applicable experimental data set such that there is a 95 percent probability with 95 percent confidence that DNB will not occur when the minimum DNBR is at the DNBR limit (1.17 for the WRB-1 correlation).

In meeting this design basis, uncertainties in nuclear and thermal parameters, and fuel fabrication parameters were combined statistically with the DNB correlation uncertainties to determine the plant DNBR uncertainty and establish the design DNBR limit such that there is at least a 95% probability with 95% confidence level that the minimum DNBR for the limiting fuel rod is greater than or equal to the DNBR limit. For this application, the design DNBR limit is 1.21. This DNBR value must be met in plant safety analyses using nominal values of the input parameters that were included in the DNBR uncertainty evaluation. In addition, margin has been maintained in the design by meeting a safety analysis DNBR limit of 1.33 in performing safety analyses.

The curves of Figure 2.1-1, show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature for which the minimum DNBR is no less than the safety analysis DNBR limit or the average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid.

The curves are based on an enthalpy hot channel factor, $F^N_{\Delta H}$, of 1.62 and a reference cosine with a peak of 1.55 for axial power shape. An allowance is included for an increase in $F^N_{\Delta H}$ at reduced power based on the expression:

$$F^N_{\Delta H} \leq 1.62\ [1 + 0.3\ (1-P)]$$

where P is the fraction of RATED THERMAL POWER

These limiting heat flux conditions are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion assuming the axial power imbalance is within the limits of the f(ΔI) function of the Overtemperature trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature ΔT trip will reduce the setpoint to provide protection consistent with core safety limits.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor pressure vessel and pressurizer are designed to Section III of the ASME Code for Nuclear Power Plant which permits a maximum transient pressure of 110% (2735 psig) of design pressure. The Reactor Coolant System piping and fittings are designed to ANSI B 31.1 and the valves are designed to ASA 16.5 which permit a maximum transient pressure of 120% (2985) psig of component design pressure. The Safety Limit of 2735 psig is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System is hydrotested at 3107 psig to demonstrate integrity prior to initial operation.

The Power Range Negative Rate trip provides protection to ensure that the minimum DNBR is maintained above the design DNBR limit for control rod drop accidents. At high power a single or multiple rod drop accident could cause flux peaking which, when in conjunction with nuclear power being maintained equivalent to turbine power by action of the automatic rod control system, could cause an unconservative local DNBR to exist. The Power Range Negative Rate trip will prevent this from occurring by tripping the reactor. For those transients on which reactor trip on power range negative rate trip is not postulated, it is shown that the minimum DNBR is greater than the design DNBR limit.

Intermediate and Source Range, Nuclear Flux

The Intermediate and Source Range, Nuclear Flux trips provide reactor core protection during reactor start-up. These trips provide redundant protection to the low setpoint trip of the Power Range, Neutron Flux channels. The Source Range Channels will initiate a reactor trip at about 10^{+5} counts per second unless manually blocked when P-6 becomes active. The Intermediate Range Channels will initiate a reactor trip at a current level proportional to approximately 25 percent of RATED THERMAL POWER unless manually blocked when P-10 becomes active. No credit was taken for operation of the trips associated with either the Intermediate or Source Range Channels in the accident analyses; however, their functional capability at the specified trip settings is required by this specification to enhance the overall reliability of the Reactor Protection System.

Overtemperature Δ T

The Overtemperature Δ T trip provides core protection to prevent DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided that the transient is slow with respect to piping transit delays from the core to the temperature detectors (about 4 seconds), and pressure is within the range between the High and Low Pressure reactor trips. This setpoint includes corrections for changes in density and heat capacity of water with temperature and dynamic compensation for piping delays from the core to the loop temperature detectors. With normal axial power distribution, this reactor trip limit is always below the core safety limit as shown on Figures 2.1-1, 2.1-2 and 2.1-3. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the reactor trip is automatically reduced according to the notations in Table 2.2-1.

through the pressurizer safety valves. No credit was taken for operation of this trip in the accident analyses; however, its functional capability at the specified trip setting is required by this specification to enhance the overall reliability of the Reactor Protection System.

Loss of Flow

The loss of Flow trips provide core protection to prevent DNB in the event of a loss of one or more reactor coolant pumps.

Above 11 percent of RATED THERMAL POWER, an automatic reactor trip will occur if the flow in any two loops drop below 90% of nominal full loop flow. Above 31% (P-8) of RATED THERMAL POWER, automatic reactor trip will occur if the flow in any single loop drops below 90% of nominal full loop flow. This latter trip will prevent the minimum value of the DNBR from going below the design DNBR limit during normal operational transients and anticipated transients when 2 loops are in operation and the Overtemperature ΔT trip setpoint is adjusted to the value specified for all loops in operation. With the Overtemperature ΔT trip setpoint adjusted to the value specified for 2 loop operation, the P-8 trip at 66% RATED THERMAL POWER with loop stop valves open and at 71% RATED THERMAL POWER with a loop stop valve closed will prevent the minimum value of the DNBR from going below the design DNBR limit during normal operational transients and anticipated transients with 2 loops in operation.

Steam Generator Water Level

The Steam Generator Water Level Low-Low trip provides core protection by preventing operation with the steam generator water level below the minimum volume required for adequate heat removal capacity. The specified setpoint provides allowance that there will be sufficient water inventory in the steam generators at the time of trip to allow for starting delays of the auxiliary feedwater system.

Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level

The Steam/Feedwater Flow Mismatch in coincidence with a Steam Generator Low Water Level trip is not used in the transient and accident analyses but is included in Table 2.2-1 to ensure the functional capability of the specified trip settings and thereby enhance the overall

ROD DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual full length (shutdown and control) rod drop time from the fully withdrawn position shall be ≤ 2.7 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

a. $T_{avg} \geq 541°F$, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODE 3.

ACTION:

a. With the drop time of any full length rod determined to exceed the above limit, restore the rod drop time to within the above limit prior to proceeding to MODE 1 or 2.

b. With the rod drop times within limits but determined with 2 reactor coolant pumps operating, operation may proceed provided THERMAL POWER is restricted to:

1. ≤ 61% of RATED THERMAL POWER when the reactor coolant stop valves in the nonoperating loop are open, or

2. ≤ 66% of RATED THERMAL POWER when the reactor coolant stop valves in the nonoperating loop are closed.

SURVEILLANCE REQUIREMENTS

4.1.3.4 The rod drop time of full length rods shall be demonstrated through measurement prior to reactor criticality:

a. For all rods following each removal of the reactor vessel head.

b. For specifically affected individual rods following any maintenance on or modification to the control rod drive system which could affect the drop time of those specific rods, and

c. At least once per 18 months.

3.2.2 $F_Q(Z)$ shall be limited by the following relationships:

$$F_Q(Z) \leq \frac{2.40}{P} [K(Z)] \text{ for } P > 0.5$$

$$F_Q(Z) \leq [4.80] [K(Z)] \text{ for } P \leq 0.5$$

where $P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$

and K(Z) is the function obtained from Figure 3.2-2 for a given core height location.

APPLICABILITY: Mode 1

ACTION:

With $F_Q(Z)$ exceeding its limit:

a. Reduce THERMAL POWER at least 1% for each 1% $F_Q(Z)$ exceeds the limit within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower Δ T Trip Setpoints have been reduced at least 1% for each 1% $F_Q(Z)$ exceeds the limit. The Overpower ΔT Trip Setpoint reduction shall be performed with the reactor subcritical.

b. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; THERMAL POWER may then be increased provided $F_Q(Z)$ is demonstrated through incore mapping to be within its limit.

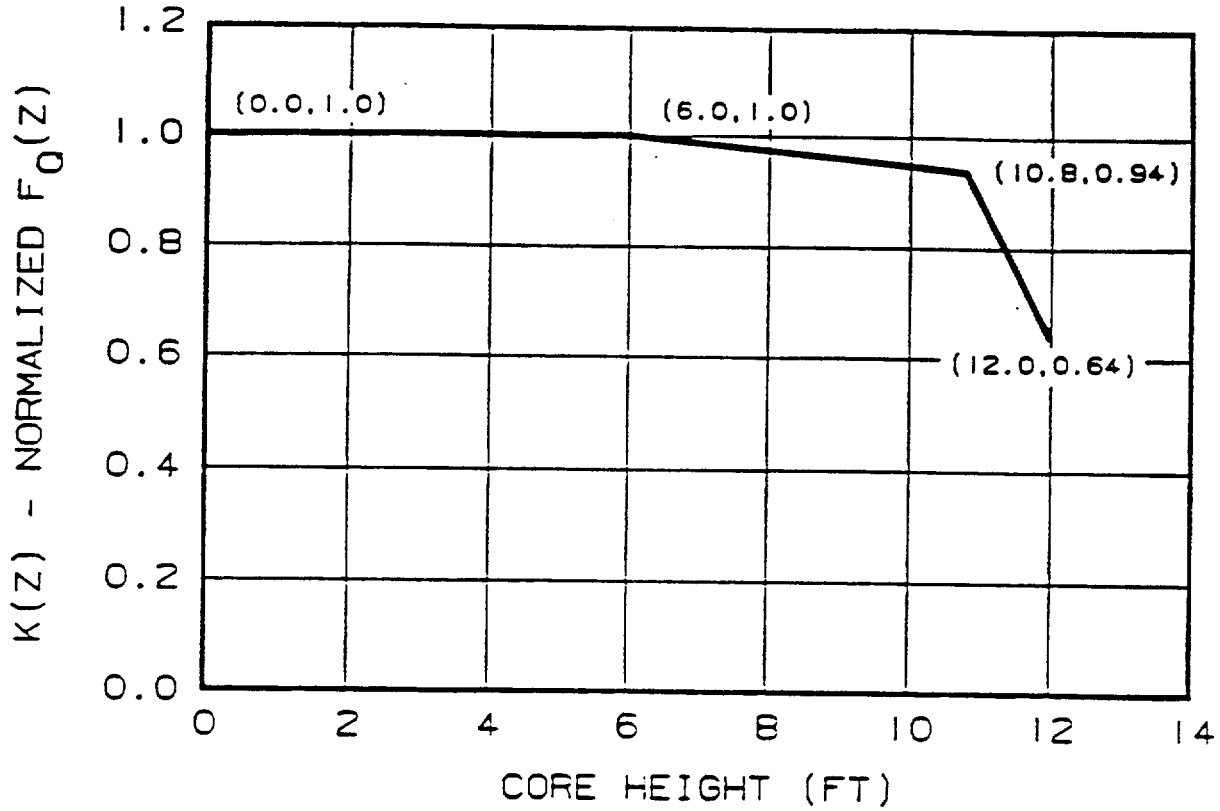

Figure 3.2-2. K(Z) - Normalized $F_Q(Z)$ as a Function of Core Height

NUCLEAR ENTHALPY HOT CHANNEL FACTOR - $F^N_{\Delta H}$
LIMITING CONDITION FOR OPERATION

3.2.3 $F^N_{\Delta H}$ shall be limited by the following relationship:

$$F^N_{\Delta H} \leq 1.62\ [1 + 0.3\ (1-P)]$$

where $P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$

APPLICABILITY: MODE 1

ACTION:

With $F^N_{\Delta H}$ exceeding its limit:

a. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to ≤ 55% of RATED THERMAL POWER within the next 4 hours.

b. Demonstrate thru in-core mapping that $F^N_{\Delta H}$ is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours, and

c. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER, subsequent POWER OPERATION may proceed provided that $F^N_{\Delta H}$ is demonstrated through in-core mapping to be within its limit at a nominal 50% of RATED THERMAL POWER prior to exceeding this THERMAL power, at a nominal 75% of RATED THERMAL POWER prior to exceeding this THERMAL power and within 24 hours after attaining 95% or greater RATED THERMAL POWER.

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (a) maintaining the minimum DNBR in the core ≥ the design DNBR limit during normal operation and in short term transients, and (b) limiting the fission gas release, fuel pellet temperature and cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F is not exceeded.

The definitions of hot channel factors as used in these specifications are as follows:

$F_Q(Z)$ Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods.

$F^N_{\Delta H}$ Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

The limits on AXIAL FLUX DIFFERENCE assure that the $F_Q(Z)$ upper bound envelope of 2.40 times the normalized axial peaking factor is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

Target flux difference is determined at equilibrium xenon conditions. The full length rods may be positioned within the core in accordance with their respective insertion limits and should be inserted near their normal position for steady state operation at high power levels. The value of the target flux difference obtained under these conditions divided by the fraction of RATED THERMAL POWER is the target flux difference at RATED THERMAL POWER for the associated core burnup conditions. Target flux differences for other THERMAL POWER levels are

3/4.2.2 and 3/4.2.3 HEAT FLUX AND NUCLEAR ENTHALPY HOT CHANNEL FACTORS - $F_Q(Z)$ and $F^N_{\Delta H}$

The limits on heat flux and nuclear enthalpy hot channel factors ensure that 1) the design limits on peak local power density and minimum DNBR are not exceeded 2) in the event of a LOCA the peak fuel clad temperature will not exceed the ECCS acceptance criteria limit of 2200°F.

Each of these hot channel factors are measurable but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to insure that the hot channel factor limits are maintained provided:

a. Control rod in a single group move together with no individual rod insertion differing by more than ± 12 steps from the group demand position.

b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.5.

c. The control rod insertion limits of Specifications 3.1.3.4 and 3.1.3.5 are maintained.

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE is maintained within the limits.

The relaxation in $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits. $F^N_{\Delta H}$ will be maintained within its limits provided conditions a thru d above, are maintained.

When an F_Q measurement is taken, both experimental error and manufacturing tolerance must be allowed for. 5% is the appropriate experimental error allowance for a full core map taken with the incore detector flux mapping system and 3% is the appropriate allowance for manufacturing tolerance.

The specified limit of $F^N_{\Delta H}$ contains an 8% allowance for uncertainties which means that normal, full power, three loop operation will result in $F^N_{\Delta H} \leq 1.62/1.08$.

Fuel rod bowing reduces the value of DNB ratio. Margin has been maintained between the DNBR value used in the safety analyses (1.33) and the design limit (1.21) to offset the rod bow penalty and other penalties which may apply.

The radial peaking factor F_{xy} (Z) is measured periodically to provide assurance that the hot channel factor, $F_Q(Z)$, remains within its limits. The F_{xy} limit for Rated Thermal Power F_{xy}^{RTP} as provided in the Radial Peaking Factor Limit Report per specification 6.9.1.14 was determined from expected power control maneuvers over the full range of burnup conditions in the core.

3/4.2.4 QUADRANT POWER TILT RATIO

The quadrant power tilt ratio limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during start-up testing and periodically during power operation.

The limit of 1.02 at which corrective action is required provides DNB and linear heat generation rate protection with x-y plane power tilts.

The two-hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned rod. In the event such action does not correct the tilt, the margin for uncertainty on F_Q is reinstated by reducing the maximum allowed power by 3 percent for each percent of tilt in excess of 1.0

3/4.2.5 DNB PARAMETERS

The limits on the DNB related parameters assure that each of the parameters are maintained within the normal steady state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR $\geq$ the design DNBR limit throughout each analyzed transient.

The 12 hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation. The 18 month periodic measurement of the RCS total flow rate is adequate to detect flow degradation and ensure correlation of the flow indication channels with measured flow such that the indicated percent flow will provide sufficient verification of flow rate on a 12 hour basis.

3/4.4.1 REACTOR COOLANT LOOPS

The plant is designed to operate with all reactor coolant loops in operation and maintain DNBR above the design DNBR limit during all normal operations and anticipated transients. In Modes 1 and 2, with one reactor coolant loop not in operation, THERMAL POWER is restricted to ≤ 31 percent of RATED THERMAL POWER until the Overtemperature ΔT trip is reset. Either action ensures that the DNBR will be maintained above the design DNBR limit. A loss of flow in two loops will cause a reactor trip if operating above P-7 (11 percent of RATED THERMAL POWER) while a loss of flow in one loop will cause a reactor trip if operating above P-8 (31 percent of RATED THERMAL POWER).

In MODE 3, a single reactor coolant loop provides sufficient heat removal capability for removing decay heat; however, due to the initial conditions assumed in the analysis for the control rod bank withdrawal from a subcritical condition, two operating coolant loops are required to meet the DNB design basis for this Condition II event.

In MODES 4 and 5, a single reactor coolant loop or RHR subsystem provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops be OPERABLE. Thus, if the reactor coolant loops are not OPERABLE, this specification requires two RHR loops to be OPERABLE.

The operation of one Reactor Coolant Pump or one RHR pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reduction will, therefore, be within the capability of operator recognition and control.

The restrictions on starting a Reactor Coolant Pump with one or more RCS cold legs less than or equal to 275°F are provided to prevent RCS pressure transients, caused by energy additions from the secondary system, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits.

The results of the spent fuel pool criticality analysis (August 1986) for Westinghouse STD/Vantage 5H and OFA/Vantage 5 fuel in three of four storage locations show that there is more than 0.3% margin to the k_{eff} limit of 0.95 with all uncertainties included. Based on the sensitivity study completed with this analysis, an increase in the maximum allowed enrichment for fuel stored in the spent fuel storage racks from 4.00 to 4.05 w/o will increase the maximum rack k_{eff} by less than 0.002. Therefore, with Westinghouse 17 x 17 STD/Vantage 5H and OFA/Vantage 5 fuel enriched at 4.05 w/o stored in the spent fuel racks in three of four storage locations and with all of the assumptions and conservatisms presented in the criticality analysis, the maximum rack k_{eff} will be less than 0.95.

3/4.9.15 CONTROL ROOM EMERGENCY HABITABILITY SYSTEMS

The OPERABILITY of the control room emergency habitability system ensures that the control room will remain habitable for operations personnel during and following all credible accident conditions. The ambient air temperature is controlled to prevent exceeding the allowable equipment qualification temperature for the equipment and instrumentation in the control room. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rem or less whole body, or its equivalent. This limitation is consistent with the requirements of General Design Criteria 19 of Appendix "A", 10 CFR 50.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 144 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

1.0 BACKGROUND

By letter dated May 9, 1989 (Reference 1), supplemented by letter dated July 21, 1989 (Reference 9), Duquesne Light Company (the licensee, acting as agent for the above utilities) proposed Technical Specification (TS) changes to support operation of the Beaver Valley Unit 1, Cycle 8 reload core and future cores with the VANTAGE 5H fuel design. The VANTAGE 5H fuel design evolved from the VANTAGE 5, Optimized Fuel Assembly (OFA), and Standard (STD) fuel assembly designs. The features of the VANTAGE 5H fuel assembly include Zircaloy Grids, Reconstitutable Top Nozzles, Debris Filter Bottom Nozzles (DFBNs), Snag Resistant Grids and Standardized Fuel Pellets. In addition, the VANTAGE 5H fuel uses Integral Fuel Burnable Absorber and Axial Blanket design features. The Technical Specification changes include (1) use of Westinghouse WRB-1 DNBR correlation for VANTAGE 5H fuel, (2) an increased control rod (RCCA) drop time from 2.2 to 2.7 seconds, (3) an increased maximum F^N_H from 1.55 to 1.62 and (4) an increased maximum F_Q from 2.32 to 2.40.

The VANTAGE 5H features were previously reviewed and approved by NRC in the Westinghouse topical report WCAP-10444-P-A, "Reference Core Report VANTAGE 5 Fuel Assembly," Addendum 2 (Reference 2). During the review of VANTAGE 5 fuel design described in WCAP-10444-P-A, the staff identified a few conditions to be resolved for those licensees using VANTAGE 5 fuel design. Since the VANTAGE 5H fuel design adopts some features from the VANTAGE 5 fuel design, our review of the Beaver Valley Unit 1 Cycle 8 reload with the VANTAGE 5H fuel design and the associated TS changes (References 3 through 8) will address those conditions listed in the safety evaluation of WCAP-10444-P-A that affect the Beaver Valley VANTAGE 5H fuel.

2.0 EVALUATION

2.1 Statistical Convolution

In the SER on WCAP-10444-P-A, we stated that the statistical convolution method should not be used in VANTAGE 5 for evaluating the fuel rod shoulder gap. The licensee (Reference 9) indicated that the statistical convolution method was not used for the VANTAGE 5H fuel design and the currently approved method was used for evaluating rod shoulder gap. Therefore, we conclude this approach is acceptable.

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P PDC

2.2 Irradiation Demonstration Program

In the SER on WCAP-10444-P-A, we required that an irradiation program be performed to confirm the VANTAGE 5 fuel performance. The licensee indicated that there were numerous demonstration programs involving OFA fuel assemblies containing Zircaloy grids irradiated into 14x14, 15x15, and 17x17 cores. The satisfactory performance of these demonstration assemblies resulted in OFA with Zircaloy grids reload applications in many Westinghouse reactors. The OFA fuel assemblies with Zircaloy grids cover the VANTAGE 5H fuel design features. Thus, we conclude that the VANTAGE 5H assemblies will perform satisfactorily in the Beaver Valley Unit 1 core.

2.3 MINI - Revised Thermal Design Procedure (MINI RTDP)

The licensee used the design procedure, called MINI RTDP (Reference 10), to perform the DNBR analysis of the core containing both 17x17 STD and VANTAGE 5H fuel assemblies. The procedure, MINI RTDP, is a revised version of the NRC-approved Westinghouse Revised Thermal Design Procedure (RTDP) (Reference 11). In RTDP, uncertainties in plant operating parameters are statistically treated such that there is at least a 95 percent probability at a 95 percent confidence level that the minimum DNBR will be greater than the design safety limit. MINI RTDP excludes plant uncertainties on reactor power, flow, temperature, pressure and bypass flow from the statistical combination process. Instead, these uncertainties will be used to offset the nominal values of these parameters for transient analyses resulting in more adverse initial conditions, as is done in older plants which do not use the statistical combination process. This method was previously reviewed and approved (Ref. 12) by NRC for the Westinghouse core reload application and, therefore, is acceptable for the Beaver Valley Cycle 8 reload calculatons.

2.4 Transient Analysis

In the SER on WCAP-10444-P-A, we required that plant specific-analysis be performed to show that the design DNBR safety limit is not violated with the higher value of F^N_H. The licensee evaluated all the transient analyses related to the DNBR calculations for Beaver Valley Unit 1 upgraded to VANTAGE 5H fuel, and plant operation with increased maximum F^N_H (from 1.55 to 1.62), increased maximum F_Q (from 2.32 to 2.4) and increased RCCA drop time (from 2.2 to 2.7 seconds). The licensee also assumed the steam generator tube plugging to a level of 10% in the evaluation. The transients evaluated by the licensee are as follows.

1. Overtemperature and Overpower Delta-T Protection (FSAR Appendix 14D).

2. Increase in Heat Removal by the Secondary System
   a. Excess Heat Removal due to Feedwater System Malfunction (FSAR 14.1.9),
   b. Excess Load Increase Incident (FSAR 14.1.10),
   c. Accidental Depressurization of the Main Steam System (FSAR 14.1.13) and Steamline Ruptures (FSAR 14.2.11).

3. Decrease in Heat Removal by the Secondary System
   a. Loss of External Load and/or Turbine Trip (FSAR 14.1.7),
   b. Loss of normal Feedwater (FSAR 14.1.8),
   c. Loss of Offsite Power to the Station Auxiliaries (FSAR 14.1.11),
   d. Major Rupture of a Main Feedwater Line (FSAR 14.2.5.2).

4. Decrease in RCS Flow Rate
   a. Partial and Complete Loss of Forced Reactor Coolant Flow (FSAR 14.1.5 and 14.2.9),
   b. Single Reactor Coolant Pump Locked Rotor (FSAR 14.2.7).

5. Reactivity and Power Distribution Anomalies
   a. Uncontrolled RCCA Withdrawal from a Subcritical Condition (FSAR 14.1.1),
   b. Rod Cluster Control Assembly Misoperation (FSAR 14.1.3, 14.2.10),
   c. Startup of an Inactive Reactor Coolant Loop at an Incorrect Temperature (FSAR 14.1.6).
   d. Uncontrolled Boron Dilution (FSAR 14.1.4),
   e. Rupture of a Control Rod Drive Mechanism Housing (FSAR 14.2.6),
   f. Inadvertent Loading and Operation with a Fuel Assembly in Improper Position (FSAR 14.2.8).

6. Increase in Reactor Coolant Inventory
   a. Spurious Operation of the Safety Injection System at Power (FSAR 14.1.16).

7. Decrease in Reactor Coolant Inventory
   a. Accidental Depressurization of the RCS (FSAR 14.1.15).

8. Steam Line Break Mass and Energy Releases for Postulated Ruptures Inside and Outside Containment (FSAR 14.2.5).

The licensee determined the events affected significantly by the fuel design update and operating condition changes, and reanalyzed those events. In Reference 6, the licensee presented the reanalyzed results for the transients to support the reload application and Technical Specification changes (Reference 3). In Reference 8, the licensee also modified the affected FSAR pages to reflect the analytical results.

The reanalyzed events can be summarized into three categories:

1. DNBR transients affected by increase of F_H^N. The events are partial Loss of Flow, Complete Loss of Flow, RCP underfrequency, RCP Locked Rotor and Startup of an Inactive Loop at an Incorrect Temperature.

2. The transients affected by increase of F_Q. The transients are RCP Locked Rotor and Rod Ejection.

3. Fast transient affected by increase of RCCA drop time. The events are Loss of Forced Reactor Coolant Flow Locked Rotor, RCCA Bank Withdrawal from Subcritical and Rod ejection.

Since the licensee used the NRC-approved Westinghouse WRB-1 correlation (Reference 13) to calculate transient DNBRs for both the STD and VANTAGE 5H fuel and demonstrated that all applicable transient analysis acceptance criteria will not be violated for the proposed Cycle 8 core, we conclude that the transient analyses are acceptable. We also approve the revision to the updated FSAR since it is consistent with the analytical results for the Beaver Valley Unit 1 Cycle 8 core.

2.5 Reactor Coolant Pump Shaft Seizure

In the SER on WCAP-10444-P-A, we stated that the mechanistic approach in determining the fraction of fuel failures during the reactor coolant pump seizure accident was unacceptable, the fuel failure criterion should be the 95/95 DNBR limit. The licensee reanalyzed the reactor coolant pump shaft (locked rotor) accident based on a failure criterion of peak clad temperature of 2700°F. The licensee concluded that there is no fuel failure and the coolability was maintained since the calculated peak clad temperature (1875°F) remained much less than 2700°F and the amount of Zirconium-water reaction was small. As indicated above, we disapprove of the use of a mechanistic approach based on 2700°F peak clad temperature in determining the fuel failure. We requested the licensee to modify the fuel failure criterion based on the approved 95/95 DNBR limit and perform DNBR analysis based on this criterion. In response (Reference 9), the licensee reanalyzed this event using the previously approved methods and showed that maximum fuel failure was less than 18% of the total fuel rods in the core based on the 95/95 DNBR limit. The percentage of fuel failure was used by the licensee in the radiological impact assessment. Since the acceptable fuel failure criterion of 95/95 DNBR limit is used for DNBR analysis, we conclude that the reactor coolant pump shaft seizure accident is satisfactorily addressed for VANTAGE 5H fuel.

The licensee determined that 18% of the fuel rods would experience DNB. These rods were assumed to fail and their gap radioactivity available for release to the primary coolant. In Reference 9 the licensee performed detailed calculations. The parameters used by the licensee as input to the radiological assessment model are suitably conservative (i.e., in accordance with appropriate regulatory guides and Standard Review Plan sections), and the calculated consequences are small fractions of 10 CFR 100 limits. We thus find the radiological assessment acceptable. We are aware that Westinghouse has submitted topical report WCAP-11157, and that our completion of its review may affect how future radiological assessments due to this postulated accident would be performed.

2.6 Positive Moderator Temperature Coefficient

In the SER on WCAP-10444-P-A, we stated that if a positive moderator temperature coefficient (MTC) is intended, the same positive MTC should be used in the plant-specific analysis. The licensee indicated (Reference 9) that no positive MTC was considered in the submittal. We thus consider that this condition is satisfied for VANTAGE 5H fuel in the Beaver Valley Unit 1 Cycle 8 core.

2.7 LOCA Analysis

In the SER on WCAP-10444-P-A, we required that the plant specific LOCA analysis be performed to show that the requirements of 10 CFR 50 are met. The licensee examined all LOCAs and assessed the effects of the VANTAGE 5H fuel and the plant operating conditions on the Beaver Valley Unit 1 LOCA analyses. The LOCAs evaluated are as follows.

1. Large-Break LOCA (FSAR 14.3.2),
2. Small-Break LOCA (FSAR 14.3.1),
3. Steam Generator Tube Rupture (FSAR 14.2.4),
4. Blowdown Reactor Vessel and Loop Forces (FSAR 14.3.3 and Appendix B),
5. Post-LOCA Long-Term Core Cooling, Boron Evaluation (FSAR 14.3.1),
6. Hot Leg Switchover to Prevent Boron Precipitation,
7. LOCA Containment Integrity (FSAR 14.3.4).

The licensee stated that the results of all LOCAs for the VANTAGE 5H core are bounded by the current FSAR LOCA results. We therefore conclude that this condition is satisfied for the VANTAGE 5H fuel used in Beaver Valley Unit 1.

3.0 TECHNICAL SPECIFICATION CHANGES

The proposed Technical Specification changes are related to use of VANTAGE 5H fuel, a new DNBR correlation, an increased RCCA drop time and increased peaking factors. These changes are as follows.

1. VANTAGE 5H Design - Page B 3/4 9-4
   The VANTAGE 5H fuel is added to this section of the Technical Specifications basis. Since VANTAGE 5H is acceptable for use in the Beaver Valley Unit 1 core, we conclude that the changes are acceptable.

2. New DNBR Correlation - Pages B2-1, B2-6, B3/4 2-1, B3/4 2-5, B3/4 2-6, B3/4 4-1.
   A new DNBR correlation of WRB-1 is added for VANTAGE 5H fuel design. The WRB-1 correlation and its safety limit have been previously approved for licensing applications. Thus, we conclude that these changes are acceptable.

3. Increased RCCA Drop Time - Page 3/4 1-22
   The RCCA drop time is revised from less than 2.2 seconds to be less than 2.7 seconds due to the use of the VANTAGE 5H fuel design. The licensee has taken into account the effect of the increased rod (RCCA) drop time in all related safety analyses. Thus, we conclude that this change is acceptable.

4. Increased Peaking Factors - $F^N_{\Delta H}$ Pages B2-2, 3/4 2-8, B3/4 2-4)

F_Q (Pages 3/4 2-5, B3/4 2-1, 3/4 2-7)

The maximum $F^N_{\Delta H}$ and F_Q are increased from 1.55 to 1.62 and 2.32 to 2.40, respectively. Since the changes are consistent with the assumptions used in the analyses to support the Cycle 8 reload application, we conclude that these changes are acceptable.

4.0 CONCLUSIONS

We have reviewed the licensee submittal of VANTAGE 5H fuel design, Technical Specification changes and Updated FSAR page changes for the Beaver Valley Unit 1 Cycle 8 transient core and all VANTAGE 5H cores. Based on the approved generic topical report, WCAP-10444-P-A and its Addendum 2, and plant-specific analyses, we approve the use of VANTAGE 5H fuel design, Technical Specification changes and Updated FSAR page changes for the Beaver Valley Unit 1 Cycle 8 transient core, and future cores of all VANTAGE 5H fuel.

We have 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, and (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

5.0 ENVIRONMENTAL CONSIDERATION

This amendment changes requirements with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20. We have determined that the amendment involves 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. We have previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets 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 this amendment.

6.0 REFERENCES

1. Letter from J. Sieber (Duquesne Light Company) to NRC, Beaver Valley Power Station, Unit No. 1, Docket No. 50-334, License No. DPR-66. Proposed Operating License change Request No. 162, dated May 9, 1989.

2. Letters from A. Thadani (NRC) to Wiesemann (Westinghouse), Acceptance for Referencing of Topical Report WCAP-10444-P-A, Addendum 2 - VANTAGE 5H Fuel Assembly, November 1, 1988 and Clarifications on the Safety Evaluation of the Topical Report WCAP-10444-P-A Addendum 2, January 1989.

3. Attachment A to Reference 1, Proposed Technical Specification changes.

4. Attachment B to Reference 1, Safety Analysis for Proposed Technical Specification change No. 162.

5. Attachment C to Reference 1, No Significant Hazard Evaluation for Proposed Technical Specification change No. 162.

6. Plant Safety Evaluation for Beaver Valley Power Station Unit 1, Fuel Upgrade and Increase Peaking Factors, Westinghouse Electric Corporation, dated April 1989.

7. Appendix A to Reference 5, Summary of Technical Specification changes.

8. Appendix B to Reference 5, Recommended Modifications to Beaver Valley Unit 1 FSAR.

9. Letter with attachment from J. Sieber (Duquesne Light Company) to NRC, dated July 21, 1989.

10. WCAP-12178-P, MINI Revised Thermal Design Procedure (MINI RTDP), March 1989.

11. Letter from A. Thadani (NRC) to W. Johnston (Westinghouse), Acceptance for Referencing of Licensing Topical Report WCAP-11397, Revised Thermal Design Procedure, January 17, 1989.

12. Letter from A. Thadani (NRC) to W. Johnson (Westinghouse), "Acceptance for Referencing Topical Report WCAP-12178 - MINI Revised Thermal Design Procedure," July 31, 1989.

13. WCAP-8762-P-A, New Westinghouse Correlation WRB-1 for Predicting Critical Heat Flux in Rod Bundles with Mixing Vane Grids, July 1984.

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<u>Principal Contributors</u>:

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