Docket File

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

May 1, 1992

Docket No. 50-412
Serial No. BV-92-024

Mr. J. D. Sieber, Vice President
Nuclear Group
Duquesne Light Company
Post Office Box 4
Shippingport, Pennsylvania 15077-0004

Dear Mr. Sieber:

SUBJECT: AMENDMENT NO. 46 : VANTAGE 5H FUEL - CHANGE REQUEST NO. 57 (TAC NO. M81758)

The Commission has issued the enclosed Amendment No. 46 to Facility Operating License No. NPF-73 for the Beaver Valley Power Station, Unit 2, in response to your application dated October 15, 1991, as supplemented by letters dated January 27, and February 25, 1992.

The amendment revises the Appendix A Technical Specification Limiting Condition for Operation 3.1.3.4. Specifically, the amendment increases the allowable control rod drop time from 2.2 seconds to 2.7 seconds. This change provides for the use of the VANTAGE 5H fuel design in future operating cycles. In addition, certain Bases sections have been changed to reflect the modified DNB design basis which uses the new Westinghouse correlation, WRB-1, for predicting critical heat flux and the MINI Revised Thermal Design Procedure (MINI-RTDP).

The NRC staff has not completed its evaluation of the control room operator dose which results from a projected 18% of the fuel rods being damaged (due to the increased control rod drop time) during a locked rotor accident. Since the staff has determined that the offsite consequences are acceptable and, through a preliminary review, that a generally acceptable approach has been followed by Duquesne Light Company (DLC) in the calculation of control room operator doses, DLC's calculation of the control room operator dose will be acceptable for cycle 4 only, and the amendment has been conditioned accordingly through the addition of a footnote to Limiting Condition for Operation 3.1.3.4. This footnote has been discussed with your staff. The NRC staff will not authorize use of VANTAGE 5H fuel beyond cycle 4 until review of the consequences to the control room operators is completed. DLC should provide the additional meteorological information identified in the enclosed Safety Evaluation no later than July 15, 1992, so that the staff can complete the review.

The requirements of this letter affect fewer than 10 respondents, and therefore, are not subject to Office of Management and Budget review under

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P.L. 96-511. The Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

/s/

Albert W. De Agazio, Sr. Project Manager
Project Directorate I-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 46 to NPF-73
2. Safety Evaluation

cc w/enclosures:
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cc:

Jay E. Silberg, Esquire
Shaw, Pittman, Potts & Trowbridge
2300 N Street, NW.
Washington, DC 20037

Nelson Tonet, Manager
Nuclear Safety
Duquesne Light Company
Post Office Box 4
Shippingport, Pennsylvania 15077

Commissioner Roy M. Smith
West Virginia Department of Labor
Building 3, Room 319
Capitol Complex
Charleston, West Virginia 25305

John D. Borrows
Director, Utilities Department
Public Utilities Commission
180 East Broad Street
Columbus, Ohio 43266-0573

Director, Pennsylvania Emergency
Management Agency
Post Office Box 3321
Harrisburg, Pennsylvania 17105-3321

Bureau of Radiation Protection
Pennsylvania Department of
Environmental Resources
ATTN: R. Janati
Post Office Box 2063
Harrisburg, Pennsylvania 17120

Mayor of the Borrough of
Shippingport
Post Office Box 3
Shippingport, Pennsylvania 15077

Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, Pennsylvania 19406

Resident Inspector
U.S. Nuclear Regulatory Commission
Post Office Box 181
Shippingport, Pennsylvania 15077

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

DUQUESNE LIGHT COMPANY

OHIO EDISON COMPANY

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

THE TOLEDO EDISON COMPANY

DOCKET NO. 50-412

BEAVER VALLEY POWER STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 46
License No. NPF-73

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

   A. The application for amendment by Duquesne Light Company, et al. (licensee) dated October 15, 1991, as supplemented by letters dated January 27, and February 25, 1992, 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. NPF-73 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 46 , and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto are hereby incorporated in the license. DLCO shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance, to be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

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: May 1, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 46

FACILITY OPERATING LICENSE NO. NPF-73

DOCKET NO. 50-412

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 2-3	B 2-3
B 2-4	B 2-4
B 2-5	B 2-5
B 2-6	B 2-6
B 2-7	B 2-7
-	B 2-8
3/4 1-23	3/4 1-23
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 4-1	B 3/4 4-1

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 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).

Incorporating the peaking factor uncertainties in the correlation limit results in a DNBR design limit value of 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 curve of Figure 2.1-1 shows 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 conservatively based on an enthalpy hot channel factor, $F^N_{\Delta H}$, of 1.62. The Thermal-Hydraulic and non-LOCA analyses that were conducted for Unit 1 bounds the Unit 2 analyses [i.e., $F^N_{\Delta H}$ of 1.62]. The LOCA and Core Design licensing basis is 1.55. These will bound actual plant operation which is restricted to an

2.1.1 REACTOR CORE, continued

$F^N_{\Delta H}$ limit of 1.55 (see the CORE OPERATING LIMITS REPORT). An allowance is included for an increase in $F^N_{\Delta H}$ at reduced power based on the expression provided in the Core Operating Limits Report (COLR).

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 ΔT 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, pressurizer, and the RCS piping, valves and fittings are designed to Section III of the ASME Code for Nuclear Power Plants which permits a maximum transient pressure of 110% (2735 psig) of 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.

2.2.1 REACTOR TRIP INSTRUMENTATION SETPOINTS

The Reactor Trip Setpoint Limits specified in Table 2.2-1 are the nominal values at which the Reactor Trips are set for each functional unit. The Trip Setpoints have been selected to ensure that the reactor core and Reactor Coolant System are prevented from exceeding their safety limits during normal operation and design basis anticipated operational occurrences and to assist the Engineered Safety Features Actuation System in mitigating the consequences of accidents. The setpoint for a reactor trip system or interlock function is considered to be adjusted consistent with the nominal value when the "as measured" setpoint is within the band allowed for calibration accuracy.

To accommodate the instrument drift assumed to occur between operational tests and the accuracy to which setpoints can be measured and calibrated, Allowable Values for the reactor trip setpoints have been specified in Table 2.2-1. Operation with setpoints less conservative than the Trip Setpoint but within the Allowable Value is

2.2.1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS, continued

acceptable since an allowance has been made in the safety analysis to accommodate this error. An optional provision has been included for determining the OPERABILITY of a channel when its trip setpoint is found to exceed the Allowable Value. The methodology of this option utilizes the "as measured" deviation from the specified calibration point for rack and sensor components in conjunction with a statistical combination of the other uncertainties in calibrating the instrumentation. In Equation 2.2-1, $Z + R + S \leq TA$, the interactive effects of the errors in the rack and the sensor, and the "as measured" values of the errors are considered. Z, as specified in Table 2.2-1, in percent span, is the statistical summation of errors assumed in the analysis excluding those associated with the sensor and rack drift and the accuracy of their measurement. TA or Total Allowance is the difference, in percent span, between the trip setpoint and the value used in the analysis for reactor trip. R or Rack Error is the "as measured" deviation, in percent span, for the affected channel from the specified trip setpoint. S or Sensor Drift is either the "as measured" deviation of the sensor from its calibration point or the value specified in Table 2.2-1, in percent span, from the analysis assumptions. Use of Equation 2.2-1 allows for a sensor drift factor, an increased rack drift factor, and provides a threshold value for REPORTABLE EVENTS.

The methodology to derive the trip setpoints is based upon combining all of the uncertainties in the channels. Inherent to the determination of the trip setpoints are the magnitudes of these channel uncertainties. Sensors and other instrumentation utilized in these channels are expected to be capable of operating within the allowances of these uncertainty magnitudes. Rack drift in excess of the Allowable Value exhibits the behavior that the rack has not met its allowance. Being that there is a small statistical chance that this will happen, an infrequent excessive drift is expected. Rack or sensor drift, in excess of the allowance that is more than occasional, may be indicative of more serious problems and should warrant further investigation.

Manual Reactor Trip

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

Power Range, Neutron Flux

The Power Range, Neutron Flux channel high setpoint provides reactor core protection against reactivity excursions which are too rapid to be protected by temperature and pressure protective circuitry. The low setpoint provides redundant protection in the power range for a power excursion beginning from low power. The trip

associated with the low setpoint may be manually bypassed when P-10 is active (two of the four power range channels indicate a power level of above approximately 10 percent of RATED THERMAL POWER) and is automatically reinstated when P-10 becomes inactive (three of the four channels indicate a power level below approximately 10 percent of RATED THERMAL POWER).

Power Range, Neutron Flux, High Rates

The Power Range Positive Rate trip provides protection against rapid flux increases which are characteristic of rod ejection events from any power level. Specifically, this trip complements the Power Range Neutron Flux High and Low trips to ensure that the criteria are met for rod ejection from partial power.

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 multiple rod drop accident could cause local 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. No credit is taken for operation of the Power Range Negative Rate trip for those control rod drop accidents for which DNBRs will be 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 startup to mitigate the consequences of an uncontrolled rod cluster control assembly bank withdrawal from a subcritical condition. 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. Although no explicit credit was taken for operation of the Source Range Channels in the accident analyses, operability requirements in the Technical Specifications will ensure that the Source Range Channels are available to mitigate the consequences of an inadvertent control bank withdrawal in MODES 3, 4 and 5.

Overtemperature ΔT

The Overtemperature ΔT 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, thermowell, and RTD response time 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 transport, thermowell, and RTD response time delays from the core to RTD output indication. With normal axial power distribution, this reactor trip limit is always below the core safety limit as shown on Figure 2.1-1. 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 notation in Table 2.2-1.

Overpower ΔT

The Overpower ΔT reactor trip provides assurance of fuel integrity, e.g., no melting, under all possible overpower conditions, limits the required range for Overtemperature ΔT protection, and provides a backup to the High Neutron Flux Trip. The setpoint includes corrections for changes in density and heat capacity of water with temperature, and dynamic compensation for transport, thermowell, and RTD response time delays from the core to RTD output indication. The Overpower ΔT trip provides protection to mitigate the consequences of various size steam line breaks as reported in WCAP-9226, "Reactor Core Response to Excessive Secondary Steam Release."

Pressurizer Pressure

The Pressurizer High and Low Pressure trips are provided to limit the pressure range in which reactor operation is permitted. The High Pressure trip is backed up by the pressurizer code safety valves for RCS overpressure protection, and is therefore set lower than the set pressure for these valves (2485 psig). The Low Pressure trip protects against low pressure which could lead to DNB by tripping the reactor in the event of a loss of reactor coolant pressure.

On decreasing power the Low Pressure trip is automatically blocked by P-7 (a power level of approximately 10% of RATED THERMAL POWER or turbine impulse chamber pressure at approximately 10% of full power equivalent); and on increasing power, automatically reinstated by P-7.

Pressurizer Water Level

The Pressurizer High Water Level trip ensures protection against Reactor Coolant System overpressurization by limiting the water level

to a volume sufficient to retain a steam bubble and prevent water relief through the pressurizer safety valves. On decreasing power, the pressurizer high water level trip is automatically blocked by P-7 (a power level of approximately 10 percent of RATED THERMAL POWER or a turbine impulse chamber pressure at approximately 10 percent of full power equivalent); and on increasing power, automatically reinstated by P-7. 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 10 percent of RATED THERMAL POWER, an automatic reactor trip will occur if the flow in any two loops drop below 90 percent of nominal full loop flow. Above 30 percent (P-8) of RATED THERMAL POWER, automatic reactor trip will occur if the flow in any single loop drops below 90 percent of nominal full loop flow.

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 here will be sufficient water inventory in the steam generators at the time of trip to allow for starting delays of the auxiliary feedwater system.

Undervoltage and Underfrequency - Reactor Coolant Pump Busses

The Undervoltage and Underfrequency Reactor Coolant Pump bus trips provide reactor core protection against DNB as a result of loss of voltage or underfrequency to more than one reactor coolant pump. The specified setpoints assure a reactor trip signal is generated before the low flow trip setpoint is reached. Time delays are incorporated in the underfrequency and undervoltage trips to prevent spurious reactor trips from momentary electrical power transients. For undervoltage, the delay is set so that the time required for a signal to reach the reactor trip breakers following the simultaneous trip of two or more reactor coolant pump bus circuit breakers shall not exceed 1.2 seconds. For underfrequency, the delay is set so that the time required for a signal to reach the reactor trip breakers after the underfrequency trip setpoint is reached shall not exceed 0.6 seconds.

On decreasing power, the Undervoltage and Underfrequency Reactor Coolant Pump bus trips are automatically blocked by P-7 (a power level of approximately 10 percent of RATED THERMAL POWER with a

turbine impulse chamber pressure at approximately 10 percent of full power equivalent); and on increasing power, reinstated automatically by P-7.

Turbine Trip

A Turbine Trip causes a direct reactor trip when operating above P-9. Each of the turbine trips provide turbine protection and reduce the severity of the ensuing transient. No credit was taken in the accident analyses for operation of these trips. Their functional capability at the specified trip settings is required to enhance the overall reliability of the Reactor Protection System.

Safety Injection Input from ESF

If a reactor trip has not already been generated by the reactor protective instrumentation, the ESF automatic actuation logic channels will initiate a reactor trip upon any signal which initiates a safety injection. This trip is provided to protect the core in the event of a LOCA. The ESF instrumentation channels which initiate a safety injection signal are shown in TABLE 3.3-3.

Reactor Coolant Pump Breaker Position Trip

The Reactor Coolant Pump Breaker Position Trips are anticipatory trips which provide reactor core protection against DNB resulting from the opening of two or more pump breakers above P-7. These trips are blocked below P-7. The open/close position trips assure a reactor trip signal is generated before the low flow trip setpoint is reached. No credit was taken in the accident analyses for operation of these trips. Their functional capability at the open/close position settings is required to enhance the overall reliability of the Reactor Protection System.

Reactor Trip System Interlocks

The Reactor Trip System interlocks perform the following functions:

P-6 Above the setpoint P-6 allows the manual block of the Source Range reactor trip and de-energizing of the high voltage to the detectors. Below the setpoint source range level trips are automatically reactivitated and high voltage restored.

P-7 Above the setpoint P-7 automatically enables reactor trips on low flow or coolant pump breaker open in more than one primary coolant loop, reactor coolant pump bus undervoltage and underfrequency, pressurizer low pressure and pressurizer high level. Below the setpoint the above listed trips are automatically blocked.

P-8 Above the setpoint P-8 automatically enables reactor trip on low flow in one or more primary coolant loops. Below the setpoint P-8 automatically blocks the above listed trip.

P-9 Above the setpoint P-9 automatically enables a reactor trip on turbine trip. Below the setpoint P-9 automatically blocks a reactor trip on turbine trip.

P-10 Above the setpoint P-10 allows the manual block of the Intermediate Range reactor trip and the low setpoint Power Range reactor trip; and automatically blocks the Source Range reactor trip and de-energizes the Source Range high voltage power. Below the setpoint the Intermediate Range reactor trip is automatically reactivated. Provides input to P-7.

P-13 Provides input to P-7.

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.

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.

*For cycle 4 operation only. Approval pending for operation beyond cycle 4 with this rod drop time.

SHUTDOWN ROD INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.5 All shutdown rods shall be within the insertion limits specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: MODES 1* and 2*#

ACTION:

With a maximum of one shutdown rod inserted beyond the insertion limit, except for surveillance testing pursuant to Specification 4.1.3.1.1, within one hour either:

a. Restore the rod to within the limit, or

b. Declare the rod to be inoperable and apply Specification 3.1.3.1.

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each shutdown rod shall be determined to be within the insertion limit:

a. Within 15 minutes prior to withdrawal of any rods in control banks A, B, C or D during an approach to reactor criticality, and

b. At least once per 24 hours thereafter.

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* See Special Test exception 3.10.2 and 3.10.3

\# With Keff ≥ 1.0

BASES

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 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 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 obtained by multiplying the RATED THERMAL POWER value by the appropriate fractional THERMAL POWER level. The periodic updating of the target flux difference value is necessary to reflect core burnup considerations.

Although it is intended that the plant will be operated with the AXIAL FLUX DIFFERENCE within the target band about the target flux difference, during rapid plant THERMAL POWER reductions, control rod motion will cause the AFD to deviate outside of the target band at reduced THERMAL POWER levels. This deviation will not affect the xenon redistribution sufficiently to change the envelope of peaking factors which may be reached on a subsequent return to RATED THERMAL POWER (with the AFD within the target band) provided the time

AXIAL FLUX DIFFERENCE (AFD) (Continued)

duration limit of the deviation is limited. Accordingly, a 1 hour penalty deviation limit cumulative during the previous 24 hours is provided for operation outside of the target band but within the limits specified in the CORE OPERATING LIMITS REPORT for THERMAL POWER levels between 50% and 90% of RATED THERMAL POWER. For THERMAL POWER levels between 15% and 50% of RATED THERMAL POWER, deviations of the AFD outside of the target band are less significant. The penalty of 2 hours actual time reflects this reduced significance.

Provisions for monitoring the AFD on an automatic basis are derived from the plant process computer through the AFD Monitor Alarm. The computer determines the one minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for at least 2 of 4 or 2 of 3 OPERABLE excore channels are outside the target band and the THERMAL POWER is greater than 90% of RATED THERMAL POWER. During operation at THERMAL POWER levels between 50% and 90% and between 15% and 50% RATED THERMAL POWER, the computer outputs an alarm message when the penalty deviation accumulates beyond the limits of 1 hour and 2 hours, respectively.

Figure B 3/4 2-1 shows a typical monthly target band near the beginning of core life.

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 and 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 Specification 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 rods 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 3.1.3.6.

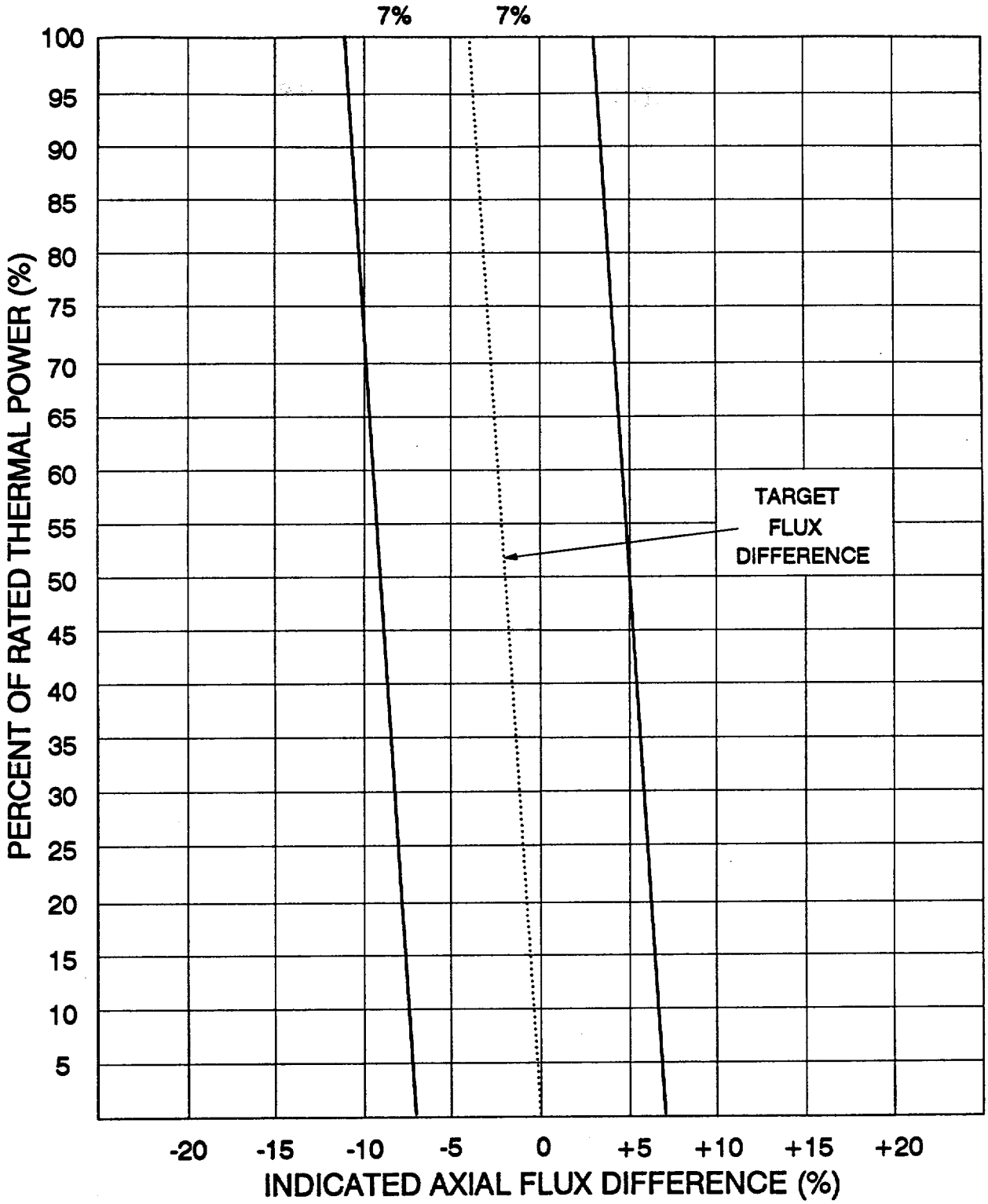

FIGURE B 3/4 2-1
TYPICAL INDICATED AXIAL FLUX DIFFERENCE (AFD) VERSUS THERMAL POWER AT BOL

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

c. The control rod insertion limits of Specifications 3.1.3.5 and 3.1.3.6 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 through 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 $F^N_{\Delta H}$ less than or equal to the design limit specified in the CORE OPERATING LIMITS REPORT.

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

The radial peaking reactor $F_{xy}(Z)$ is measured periodically to provide assurance that the hot channel factor, $F_Q(Z)$, remains within its limit. The F_{xy} limit for Rated Thermal Power (F^{RTP}_{xy}) provided in the CORE OPERATING LIMITS REPORT 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 startup 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 uncertainly 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 PARAMETER

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 greater than or equal to 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 AND COOLANT CIRCULATION

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, this specification requires that the plant be in at least HOT STANDBY within 6 hours.

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 when the rod control system is capable of control bank rod withdrawal.

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 350°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 of appendix G by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

3/4.4.2 and 3/4.4.3 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 345,000 lbs. per hour of saturated steam at the valve set point. The relief capacity of a single safety valve is adequate to 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 reactor 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 Section XI of the ASME Boiler and Pressure Code.

Safety valves similar to the pressurizer code safety valves were tested under an Electric Power Research Institute (EPRI) program to determine if the valves would operate stably under feedwater line break accident conditions. The test results indicated the need for inspection and maintenance of the safety valves to determine the potential damage that may have occurred after a safety valve has lifted and either discharged the loop seal or discharged water through the valve. Additional action statements require safety valve inspection to determine the extent of the corrective actions required to ensure the valves will be capable of performing their intended function in the future.

3/4.4.4 PRESSURIZER

The requirement that 150 Kw of pressurizer heaters and their associated controls and 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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 46 TO FACILITY OPERATING LICENSE NO. NPF-73

DUQUESNE LIGHT COMPANY

OHIO EDISON COMPANY

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

THE TOLEDO EDISON COMPANY

BEAVER VALLEY POWER STATION, UNIT 2

DOCKET NO. 50-412

1.0 INTRODUCTION

By letter dated October 15, 1991, Duquesne Light Company (DLC) proposed a change to the Beaver Valley Power Station, Unit No. 2 Appendix A Technical Specifications (TS). The proposed change would increase the allowable control rod drop time specified in Limiting Condition for Operation (LCO) 3.1.3.4 to 2.7 seconds from 2.2 seconds. This change would allow the use of the VANTAGE 5 Hybrid (VANTAGE 5H) fuel design which incorporates a smaller thimble tube diameter. The smaller thimble tube diameter results in a slightly greater rod drop time. In addition, DLC proposed changes to certain Bases sections to reflect the modified DNB design basis which uses the new Westinghouse correlation, WRB-1, for predicting critical heat flux and the MINI Revised Thermal Design Procedure (MINI-RTDP).

Additional supporting information was submitted by letters dated January 27, 1992, and February 25, 1992. The additional information did not change the initial proposed no significant hazards consideration determination.

2.0 BACKGROUND

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 VANTAGE 5H features were previously reviewed and approved by the NRC in Westinghouse topical report WCAP-10444-P-A, "Reference Core Report VANTAGE 5 Fuel Assembly, Addendum 2." During the review of VANTAGE 5 fuel design described in WCAP-10444-P-A, the staff identified conditions to be resolved for the licensees using the VANTAGE 5 fuel design. Since the VANTAGE 5H fuel

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design adopts some features from the VANTAGE 5 fuel design, the staff evaluation, provided below, of the Beaver Valley Unit 2 Cycle 4 reload with the VANTAGE 5H fuel design and the associated TS changes, addresses each of those conditions listed in the safety evaluation of WCAP-10444-P-A that affect the Beaver Valley 5H fuel. This approach is the same as that used by the staff in their review of the Beaver Valley Unit 1 Cycle 8 reload with VANTAGE 5H fuel.

3.0 EVALUATION

The conditions from WCAP-10444-P-A that were found to be applicable to Beaver Valley Unit 2 Cycle 4 reload with the VANTAGE 5H fuel design were reviewed by the staff. The conditions reviewed were Statistical Convolution, Irradiation Demonstration Programs, MINI-Revised Thermal Design Procedure, Transient Analysis, Reactor Coolant Pump Shaft Seizure accident, and Loss Of Coolant Accident (LOCA) Analysis.

3.1 Statistical Convolution

In the safety evaluation (SE) for WCAP-10444-P-A, it was stated that the statistical convolution method was not approved for evaluating the uncertainties associated with the axial fuel rod to nozzle gap for the VANTAGE 5 fuel design. In the case of VANTAGE 5H fuel, the convolution method also would also not be an approved method for evaluating fuel rod to nozzle gap. The staff's review confirmed that the convolution method was not used in the Cycle 4 analysis.

The VANTAGE 5H rod gap has been increased in Cycle 4 by the use of the Reconstitutable Top Nozzle which has a reduced nozzle plate thickness. The thickness reduction results in additional space for fuel rod growth. Section 2.1.1 of WCAP-10444-P-A indicates that the fuel rod gap growth is acceptable for the VANTAGE 5 fuel design. It is therefore acceptable for the Unit 2 Cycle 4 reload.

3.2 Irradiation Demonstration Program

The SE for WCAP-10444-P-A required that an irradiation demonstration program be performed to confirm the VANTAGE 5 fuel performance. DLC has done this in that the evaluation of the VANTAGE 5H grid performance is based on: (1) the extensive design and irradiation experience with previous grid designs, and (2) full grid testing completed with the VANTAGE 5H grid design. DLC indicated that there were fuel assembly demonstration programs that entailed inserting optimized fuel assemblies (OFA) containing Zircaloy grids into 14x14, 15x15, and 17x17 cores. The satisfactory performance of these demonstration assemblies resulted in OFAs with Zircaloy grids being used in reloads and operating successfully since the early 1980s in many Westinghouse cores.

3.3 MINI - Revised Thermal Design Procedure (MINI RTDP)

The existing thermal-hydraulic analysis of the 17x17 STD fuel used in the Beaver Valley Unit 2 plant is based on standard thermal and hydraulic methods and the W-3 (R-Grid) DNB correlation as described in the Beaver Valley Unit 2 Updated Final Safety Anaslysis Report (UFSAR). The DNB analysis for the mixed core (17x17 STD and VANTAGE 5H fuel assemblies) has been modified to incorporate the WRB-1 DNB correlation and a conservative application of the Revised Thermal Design Procedure (MINI-RTDP).

With MINI-RTDP methodology, peaking factor uncertainties are combined statistically with the DNB correlation uncertainties to obtain the overall DNBR uncertainty factor. The uncertainty factor is then used to define the design limit DNBR that satisfies the DNB design criterion. This criterion states that there is at least a 95% probability at a 95% confidence level that DNB will not occur on the most limiting fuel rod for any Condition I or II event. MINI-RTDP excludes the uncertainties on primary system parameters (reactor power, flow, temperature and pressure) from the statistical combination process. These uncertainties will be used to offset the nominal values of these parameters in transient analyses, resulting in more adverse initial conditions.

The MINI-RTDP methodology was previously reviewed and approved by the NRC for the Westinghouse core reload application and, therefore, is acceptable for the Beaver Valley Cycle 4 reload calculations.

3.4 Transient Analysis

The reanalysis and evaluation of non-LOCA transients summarized below takes into consideration the effects of upgrading to VANTAGE 5H fuel and the deletion of thimble plugs. The evaluation bounds the case where some or all of the thimble plugs are present and supports up to 20% steam generator tube plugging.

The major effect of changing from STD 17x17 fuel to VANTAGE 5H fuel is the increased design Rod Control Cluster Assembly (RCCS) drop time. The slower drop time (from 2.2 to 2.7 seconds) is due to the decrease in the inner diameter of the VANTAGE 5H fuel thimble tube, as compared to the STD 17x17 fuel thimble, by 0.008 inches.

The impact of removing the thimble plugs is an increase in the core bypass flow from 4.5% to 6.5%. This increase in bypass flow results in an equivalent decrease in the flow through the core active fuel region.

The DNB limited events have either been re-analyzed incorporating the decrease in core flow and increase in rod drop time, or have been evaluated such that the results of previous analyses remain valid. In addition, the reactor core thermal limit curves have not changed and the current Technical Specification Overtemperature ΔT and Overpower ΔT setpoints remain valid.

For the events that are not DNB related, or for which the prevention of DNB is not the only safety criterion, the effects of decreased core flow and increased rod drop time have been evaluated with respect to the applicable acceptance criteria; e.g. core exit coolant temperature. The evaluation demonstrated that acceptance criteria continue to be met.

DLC has reanalyzed those events that are sensitive to increased design RCCA drop time. The events re-analyzed include loss of forced reactor coolant flow, locked rotor, control rod bank withdrawal from subcritical, and rod ejection. The WRB-1 correlation and the MINI-RTDP methodology were used to evaluate transient DNBRs for both the STD and VANTAGE 5H fuels. The transient reanalyses and evaluations demonstrate that the applicable safety analysis acceptance criterion continue to be met for the intended fuel design and are therefore acceptable.

3.5 Reactor Coolant Pump Shaft Seizure

The evaluation for locked rotor event is similar to the evaluation on Locked Rotor for Beaver Valley Unit 1. DLC has evaluated the reactor coolant pump shaft (locked rotor) accident based on the failure of the peak cladding temperature of 2700 degrees F. DLC has concluded that there is no fuel failure and the cooling was maintained since the calculated peak clad temperature (1870 degrees F) remained much less than 2700 degrees F and the amount of Zirconium-water reaction was small.

DLC found that 18% of the fuel rods could experience DNB with minimum DNBRs less than the safety analysis DNBR limit. This was calculated based on a fuel rod power census which is conservative for Cycle 4 operations and is expected to bound future cycles. It was concluded that the integrity of the primary coolant system is not endangered and the core remains intact with no consequential loss of core cooling capability.

The amount of fuel failure is used to assess the radiological consequences of the Locked Rotor event. Since the acceptable fuel failure criterion of 95/95 DNBR limit is used for DNBR analysis, it is concluded that the reactor coolant pump shaft seizure accident is satisfactorily addressed for VANTAGE 5H fuel.

3.5.1 Radiological Consequences of a Reactor Coolant Pump Shaft Seizure

DLC performed calculations to demonstrate that the offsite dose consequences of a locked rotor accident for the new fuel would still be within the NRC Standard Review Plan acceptance criteria (25% of 10 CFR Part 100 dose reference values) and that the control room operator doses would still meet the requirements of General Design Criterion (GDC) 19 of 10 CFR Part 50, Appendix A. The October 15, 1991 submittal indicated that the initial recalculation of the control room operator doses resulting from the locked-rotor event showed doses exceeding GDC 19. However, a subsequent evaluation, dated January 27, 1992, performed by DLC showed that the control room

operator doses are within the dose limits of GDC 19. This analysis of the control room operator doses relies upon a modified analysis of the atmospheric diffusion associated with the various release points at Beaver Valley 2. On February 25, 1992, DLC submitted a report prepared by Halliburton NUS Environmental Corporation which provides the basis for the revised meteorological parameters used in the analysis. Also on February 25, 1992, DLC submitted an analysis of the offsite radiological consequences of the locked-rotor event.

The staff has completed its review of the offsite consequences of the locked-rotor event. The staff considered that radioactivity would be released from the steam generator safety valves and/or the power operated relief valves under the following conditions:

1) radioactivity in the steam and secondary coolant of the steam generators at the technical specification (TS) concentrations;

2) a primary to secondary leak rate at the TS limit;

3) primary coolant at the TS limit for radioactivity concentration; and

4) a concurrent iodine spike at the time of the locked-rotor event.

The staff has evaluated the offsite consequences based upon an accident duration of 8 hours. The staff has determined independently that the dose consequences at the exclusion area boundary (EAB) and the low population zone (LPZ) are acceptable.

The staff, however, has not completed reviewing the control room operator dose analysis. DLC's analysis for the control room utilized X/Q values that were derived using the methodology presented by J. V. Ramsdell in NUREG/CR-5055, "Atmospheric Diffusion for Control Room Habitability Assessment" and in Proceedings of the 21st DOE/NRC Air Cleaning Conference (Conference 900813, NUREG/CP-0016, Vol 2, pp 714-729, 1990) "Alternatives to Current Procedures Used to Estimate Concentration in Building Wakes." (The Proceedings contained a typographical error but Mr. Ramsdell provided corrected information by letter to Halliburton NUS).

The staff has reviewed the Halliburton NUS report and has determined that insufficient information is included to support an independent staff evaluation of the X/Q values that were derived. Therefore, additional meteorological information is required; DLC shall submit, not later than July 15, 1992, the site-specific hourly meteorological data that were used to determine the X/Q values used in the calculation of doses to the control room operators. Following submittal of this information, the staff will complete the evaluation of the revised X/Q values and the resulting control room doses and will issue a supplemental safety evaluation presenting the results of their conclusions.

However, based upon a preliminary review, the staff has concluded that the approach DLC used to calculate the doses to the control room operators is generally-acceptable. Therefore, DLC's evaluation of the radiological consequences to the control room operators is accepted for Cycle 4 only.

3.6 LOCA Analysis

The LOCA analysis was evaluated to determine whether the thimble plug removal and the VANTAGE 5H low pressure drop Zircaloy grid fuel feature had an effect on the results.

The following LOCAs were evaluated:

15.6.5	Large Break LOCA
15.6.5	Small Break LOCA
15.6.3	Steam Generator Tube Failure
3.9N	Blowdown Reactor Vessel and Loop Forces
15.6.5	Post LOCA Long-Term Cooling, Subcriticality Evaluation
6.3.2.5	Hot Leg Switchover to Prevent Potential Boron Precipitation/Long-Term SI Verification (UFSAR Table 6.3-7)
15.3.4	LOCA Containment Integrity

DLC indicated that in most cases the Cycle 4 modifications are supported by the existing licensing basis safety analyses. In these cases it was concluded that specific safety analyses are not sensitive to the fuel and thimble plug removal upgrades, or have otherwise incorporated bounding analyses assumptions. In all cases, the LOCA evaluation and reanalysis demonstrate that the applicable safety criteria are met and, therefore, the staff finds the evaluation acceptable.

3.7 Overall

The NRC staff has reviewed DLC's proposed Technical Specification change to support operation of Beaver Valley Unit 2 with the VANTAGE 5H fuel design. Based on the approved generic topical reports and plant specific analysis, the NRC staff finds the use of the VANTAGE 5H fuel design acceptable for cycle 4 only. The staff will evaluate the acceptability of operation beyond cycle 4 based upon review of the additional meteorological information requested. The staff's findings will be presented in a supplemental safety evaluation. Therefore, the proposed change to TS LCO 3.1.3.4, which would increase the allowable control rod drop to 2.7 seconds from 2.2 seconds, is acceptable for cycle 4 only, and a footnote noting this limitation has been added to TS page 3/4 1-23. The footnote also notes that approval for operation with the stated allowable control rod drop time beyond cycle 4 is pending. This footnote has been discussed with a DLC representative, and it is acceptable with DLC. DLC should provide the requested meteorological information no later than July 15, 1992.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to 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 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 radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding (57 FR 2592). Accordingly, the 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 the amendment.

6.0 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 amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: S. Brewer
J. Hayes

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