Duquesne Light Company
Beaver Valley Power Station
P.O. Box 4
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October 11, 1993

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Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: Beaver Valley Power Station, Unit No. 1
Docket No. 50-334, License No. DPR-66
Response to Information Request Dated June 24, 1993
30 Percent Steam Generator Tube Plugging Analysis

This letter provides additional information in response to your request dated June 24, 1993, concerning Duquesne Light Company's 30 percent steam generator tube plugging analysis. By letters dated July 13, 1993, and August 11, 1993, we provided responses to your request by submitting the 10 CFR 50.59 safety evaluation and the initial issue of WCAP-13707, "30% Steam Generator Tube Plugging Analysis Program Final Engineering and Licensing Report." Provided in Attachment A are the items which complete our response to your request including Revision 1 of the WCAP-13707 proprietary and WCAP-13811 non-proprietary versions along with a copy of the associated proprietary information notice, copyright notice and affidavit. Also included is a copy of Revision 1 of the 10 CFR 50.59 safety evaluation.

WCAP-13707 Revision 1 contains information which is proprietary to Westinghouse Electric Corporation. Accordingly, we request that this information be withheld from public disclosure in accordance with 10 CFR 2.790.

If you have any questions regarding the attached response, please contact Mr. Steve Sovick at (412) 393-5211.

Sincerely,

J. D. Sieber

J. D. Sieber

cc: Mr. L. W. Rossbach, Sr. Resident Inspector
Mr. T. T. Martin, NRC Region I Administrator
Mr. C. E. Edison, Senior Project Manager

Ltr, Encl 1 w/ 1 Prop
A001 Change NRC PDR 1/1 Prop 1 NP

ATTACHMENT A

Beaver Valley Power Station, Unit No. 1
SUPPORTING DOCUMENTATION FOR
30 PERCENT STEAM GENERATOR TUBE PLUGGING ANALYSIS

The following documents are provided:

Proprietary Information Notice

Copyright Notice

Affidavit

10 CFR 50.59 Safety Evaluation Revision 1

WCAP-13707 Revision 1 Proprietary

WCAP-13811 Revision 1 Non-Proprietary

Proprietary Information Notice

Transmitted herewith are proprietary and/or non-proprietary versions of documents furnished to the NRC in connection with requests for generic and/or plant-specific review and approval.

In order to conform to the requirements of 10 CFR 2.790 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) contained within parentheses located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (4)(ii)(a) through (4)(ii)(f) of the affidavit accompanying this transmittal pursuant to 10 CFR 2.790(b)(1).

Copyright Notice

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

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COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared Henry A. Sepp, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Westinghouse Electric Corporation ("Westinghouse") and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

Henry A. Sepp

Henry A. Sepp, Manager
Strategic Licensing Issues

Sworn to and subscribed
before me this 26th day
of August, 1993

Denise K Henderson

Notary Public

Notarial Seal
Denise K. Henderson, Notary Public
Monroeville Boro, Allegheny County
My Commission Expires Oct. 28, 1996
Member, Pennsylvania Association of Notaries

(1) I am Manager, Strategic Licensing Issues, in the Nuclear and Advanced Technology Divisions, of the Westinghouse Electric Corporation and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rulemaking proceedings, and am authorized to apply for its withholding on behalf of the Westinghouse Energy Systems Business Unit.

(2) I am making this Affidavit in conformance with the provisions of 10CFR Section 2.790 of the Commission's regulations and in conjunction with the Westinghouse application for withholding accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by the Westinghouse Energy Systems Business Unit in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.790 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitutes Westinghouse policy and provides the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of

Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information which is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iii) The information is being transmitted to the Commission in confidence and, under the provisions of 10CFR Section 2.790, it is to be received in confidence by the Commission.

(iv) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in WCAP-13707 Rev. 1, (Proprietary), August, 1993 for Beaver Valley Unit 1, being transmitted by the Duquesne Light Company letter and Application for Withholding Proprietary Information from Public Disclosure, to Document Control Desk to the Attention Dr. Thomas Murley. The proprietary information as submitted for use by Duquesne Light Company for Beaver Valley Unit 1 is expected to be applicable in other licensee submittals in response to certain NRC requirements for justification of plant operation with specified levels of steam generator tube plugging.

This information is part of that which will enable Westinghouse to:

(a) Provide documentation of the evaluation of increased steam generator tube plugging data.

(b) Establish the methodology for analyzing operating plant data.

(c) Establish the methodology for determining rms displacement amplitudes for core barrel and thermal shield nodes.

(d) Assist the customer to obtain NRC approval.

Further this information has substantial commercial value as follows:

(a) Westinghouse plans to sell the use of similar information to its customers for purposes of meeting NRC requirements for licensing documentation.

(b) Westinghouse can sell support and defense of the technology to its customers in the licensing process.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar methodologies and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended for developing testing and analytical methods and performing tests.

Further the deponent sayeth not.

NUCLEAR GROUP ADMINISTRATIVE MANUAL 8.18

ATTACHMENT 1 - 10 CFR 50.59 EVALUATION WORKSHEET

Plant Change or Procedure No. WCAP-13707 Revision No. 1 .

Plant Change or Procedure Title: 30 PERCENT STEAM GENERATOR TUBE PLUGGING ANALYSIS PROGRAM - ENGINEERING AND LICENSING REPORT

Unit Number: 1 .

INTRODUCTION

1. Describe the plant or procedure change, test, or experiment being evaluated and its expected effects.

This evaluation is being revised to reflect Revision 1 to the WCAP. After the initial issue of WCAP 13707, minor clarifications and editorial changes were identified during the Duquesne Light Company review. These changes have been incorporated into the WCAP and it has been reissued as Rev. 1. A review of the revised WCAP has been completed, and none of the new changes has any impact on this safety evaluation; however, it is being reissued to reflect the revision. As originally described, this change includes the following: (a) a maximum plugging level of 30 percent in any one steam generator or the plugging level that results in the reduction of the RCS loop flow rate to the thermal design flow (TDF) limit; (b) a reduction in the TDF from 88,500 gpm per loop to 87,200 gpm per loop (261,600 gpm total); and (c) the incorporation of loop flow asymmetry of up to 5% in the analyses and evaluations in which RCS flow rates are important. The reduced total TDF value of 261,600 gpm (reduced by 1.5%) represents a Technical Specification change which has been reviewed by OSC and ORC and subsequently submitted to the NRC on 2/19/93. The WCAP assumes that the thimble plugs in the reactor internals are removed.

Based on the power capability parameters identified in item 3 below, the current design transients remain bounding for increased steam generator tube plugging levels up to either the 30% or the revised TDF limit. This conclusion is valid only to the minimum steam conditions of 760 psia and 512.3 F due to component design considerations. Engineering will initiate a request to include the steam pressure value into the Precautions and Limitations section of the OM.

Revised loads have been calculated for the RCP motors. The new loads have increased due to both revised performance estimates (RCS Temperatures) and the effects of the proposed tube plugging. WCAP-13707 Section 5.3.2 addresses this concern and concludes that it does not impact the safety-related function of the motors.

The expected effects of this WCAP will be to document the analyses and evaluations needed to verify that BVPS-1 is functionally and structurally capable of continued reliable and safe operation with the change.

2. Describe why the plant or procedure is being changed.

   The previous justification (WCAP-12966 dated J ne 1991) for 20 percent tube plugging was outdated based on the number of steam generator tubes plugged during 9R. The reduction in the TDF and the incorporation of loop flow asymmetry provide analysis margin which allow a tube plugging limit of up to 30%.

3. Identify the operating parameters, design parameters, and systems affected by the change.

   The design and operating parameters for the reactor vessel were revised as a result of the revised SGTP and reduced TDF. The following are the new parameters: Normal Operating Temperature F (Inlet/Outlet)- (542.0/610.4); Zero Load Temperature 547 F.

   The power capability parameters used in the analyses reflect the level of 30% steam generator tube plugging, a thermal design flowrate reduction of 1.5%, and loop flow asymmetry of up to 5%. Since the RCS thermal design flowrate was reduced, RCS temperatures reflect the increase in delta T. Thot increased, and Tcold decreased slightly due to the lower RCS flow through the core. Tavg remains unchanged. The steam temperature, pressure, and flowrate also decreased due to the increased tube plugging, and its attendant reduction in heat transfer area.

   For the 30% analysis, steam generator conditions were assumed to remain unchanged from the existing 20% analysis, reference Part 1, item A.3. Based on current plant operating conditions, there is expected to be operating margin to accommodate the anticipated steam pressure reduction caused by the higher tube plugging level. The steam pressure reduction has been "analytically" limited to 760 psia and 512.3 F.

   The following parameters are affected:

	20% Tube Plugging TDF Maintained	30% Tube Plugging TDF Reduced.
TDF GPM/Loop	88,500	87,200
RC Temp, F		
Vessel Outlet	609.9	610.4
Vessel Average	576.2	576.2
Vessel/Core Inlet	542.5	542.0
Steam Generator		
Steam Temp, F	512.4	506.5
Steam Press psia	760	721
Steam Flow lb/hr	11.60X10E6	11.59X10E6
Tube Plugging %	20	30

   Parameters associated with the margin of safety are identified in Part 3 item A.1.

4. Identify the credible failure modes associated with the change.

   The proposed change is an analysis change which will allow plugging of Steam Generator tubes to the 30% level. This change does not introduce any new failure modes; A leaking plug or failure of the mechanical seal are credible failure modes associated with the tube plugging operation; however, this analysis is to increase the allowable level of plugged tubes, as described above, not to change any mechanical or physical operation. The effect of the increased plugging level on system parameters has been evaluated and is acceptable. There are no failure modes "per se" associated with the proposed change.

5. Provide references to the location of information used for the safety evaluation.

   WCAP-13707, rev. 1, dated August 1993, and WCAP-12966 dated June 1991
   DLW-92-410 dated 12/28/92
   BV-1 SER (October 1974) and SER Supplement 2
   BV-1 Updated FSAR Sections 5.4, 14.1, 14.2, and 14.3
   ND3NSM:5925, "Technical Specification Change Nos. 1A-208/2A-74" dated 2/18/93.
   DLW-92-378:"DLCo BVPS Unit-1 RCS Reduced Thermal Design Flow Report" 12/17/92

PART 1: EFFECT ON DESIGN BASIS (UFSAR) ACCIDENTS

A. Supporting Information

A.1. Identify the safety SYSTEMS and/or SYSTEMS important to safety affected by the change.

   The Reactor Coolant System (O.M. Chapter 6); Chemical and Volume Control (O.M. Chapter 7); RHR System (O.M.Chapter 10); Safety Injection System (O.M.Chapter 11); Feedwater System (O.M. Chapter 24); Main Steam System (O.M. Chapter 21); Auxiliary Feedwater System (O.M. Chapter 24B); and the Steam Generator Blowdown (O.M. Chapter 25) as indicated in the TDF Report and in WCAP 12966.

A.2. Discuss the effects of the change and/or the failure modes associated with the change on the probability of failure of the systems identified.

   An assessment of system integrity concluded that component integrity is expected to be maintained. Operation does not significantly impact the design transients and thus does not challenge the design of these components.

A.3. Discuss the effect of the change on the performance of the safety systems.

   New flows and pressure drops were calculated for the various flow paths within the reactor pressure vessel system. The results show that the changes in pressure drop associated with the new operating conditions are evenly distributed throughout the reactor internals, and that the total pressure drop across the internals would decrease an insignificant amount. Since the internals flow and pressure drop changes are not changed significantly by the new operating conditions, detailed calculations of the effect on core bypass flow, hydraulic lift forces, flow induced vibration and RCCA rod drop times were not necessary. Additionally, the temperature rise across the reactor vessel is bounded by the original structural analysis of the BV-1 internals. The evaluation of the Reactor Pressure Vessel System demonstrated that there would be no adverse impact on the performance of the system by the proposed reduction in TDF.

For NSSS Primary System Components, the new thermal design flow has been evaluated and determined to be acceptable. The 5% asymmetry is defined as follows: The flowrate in any loop must be greater than 82,840 gpm (5% below 87,200 gpm); the sum of the flow rates in the two lowest flow loops must be greater than 170,040 gpm: and the total of all three loop flowrates must be equal to or greater than the reduced TDF limit of 261,600 gpm. These flow limits will be incorporated into test procedures.

The RCS temperatures associated with the proposed design condition will change slightly as indicated in Table 2.1-1 of the WCAP; the proposed 30% tube plugging will not affect the existing design basis analysis as indicated in the WCAP.

Section 6.3.3 indicates that the RCP motor rotor windings temperature rise due to this change could result in an accelerated level of mechanical aging (fatigue) which could result in rotor windings failure. This will not affect the safety related function of the RCP motor. The WCAP includes precautions if the motors are operated at the 30% limit.

For the Main Steam System, the Condensate and Feedwater Systems, the Auxiliary Feedwater System, and the Steam Generator Blowdown and Sampling System, the increase in the allowable steam generator tube plugging level to the 30% limit has been determined to have no adverse effects. It is significant to note that the parameters which affect Balance of Plant (BOP) systems either do not change, or change in the favorable direction. The full power steam and feedwater mass flow decrease slightly. Reactor power remains the same. The final feedwater temperature remains at 437.5 F and the full power steam pressure remains at 760 psia. The zero load steam generator temperature remains at 547 F, and this shows that the zero load steam generator pressure will remain at the present 1020 psia value. The conclusions of this evaluation were that the evaluation performed as part of the 20% SGTP effort apply to the 30% SGTP, reduced TDF conditions.

The proposed change will not affect the performance of the safety systems.

A.4. Identify the design basis accidents in the UPDATED FINAL SAFETY ANALYSIS REPORT (UFSAR) to be reviewed for potential impact by the change.

All Chapter 14 accidents, especially "Uncontrolled Boron Dilution" (14.1.4); "Excessive Heat Removal Due to Feedwater System Malfunctions" (14.1.9); "Excessive Load Increase Incident" (14.1.10); "Loss of External Electrical Load and/or Turbine Trip" (14.1.7); "Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power" (14.1.2); "Start Up of an Inactive Reactor Coolant Loop" (14.1.6); "Spurious Operation of the Safety Injection System at Power" (14.1.16); "Accidental Depressurization of the RCS" (14.1.15); "Accidental Depressurization of the MS System" (14.1.13); "Major Secondary System Pipe Rupture" (14.2.5); "Partial Loss of Forced Reactor Coolant Flow" (14.1.5); "Complete Loss of Forced RCS Flow" (14.2.9); "Single Reactor Coolant Pump Locked Rotor" (14.2.7); "Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical Condition" (14.1.1); "Rupture of a CRDM Housing RCCA Ejection" (14.2.6); Loss of Offsite Power to the Station Auxiliaries" (14.1.11); "Steam Generator Tube Rupture" (14.2.4); and "Loss of Coolant Accident" (14.3).

A.5. Discuss how the parameters and systems, affected by the change, affect the assumptions and radiological consequences of the accident(s) identified in Part 1, A.4.

Small changes in plant operating conditions such as TDF and Steam Generator Tube Plugging (SGTP) will not significantly affect the transient statepoints used in the DNBR calculations. Hence the transient conditions used to calculate the minimum DNBRs are still valid for the reduced TDF. A decrease in the RCS flow rate potentially decreases the minimum DNBR calculations during the event. Existing conservatism in the DNB calculations bound the effect on DNB due to the 1.5% flow reduction. A reduction in TDF has an adverse effect on the core thermal limits (consequently the Overtemperature and Overpower Delta-T analysis setpoint equations) and the initial conditions assumed for all of the non-LOCA transients. The core thermal limits were reviewed and revised to account for the reduced flow. The current OT-DeltaT and OP-DeltaT analysis setpoint equations were reviewed by Westinghouse and confirmed to provide protection for the revised core limits. All non-LOCA transients were examined to determine the impact of the reduced TDF. Generic margin has been allocated to ensure that the limit continues to be met.

Because the TDF reduction is limited to approximately 1.5%, existing flow sensitivities were used to demonstrate that non-DNB safety criteria (e.g. peak clad temperature, RCS pressure) will also continue to be met. Steam generator tube plugging asymmetries lead to flow asymmetries among the reactor coolant loops. The loop with the largest amount of plugging will have the lowest reactor coolant flow. Because of the mixing in the reactor vessel lower plenum, temperature asymmetries resulting from the flow asymmetries are minimized. Section 6 of the WCAP presents details of the effect on the accident analyses listed in item A.4 and concludes that there is no impact.

The reduced TDF, steam pressure, and steam temperature result in a decrease in the initial mass in the steam generators. The combination of reducing TDF and increasing tube plugging would result in a reduction in the steam generator secondary side mass on the order of < 0.5% from the existing analysis values. Only the loss of heat sink transients (Loss of Non-Emergency AC Power, Loss of Normal Feedwater and Feedwater Line Break) are potentially impacted by this minimal decrease in initial steam generator mass. A sensitivity analysis has shown that sufficient margin exists to the limit to accomodate the reduced mass. The remaining non-LOCA transients (including all of those analyzed for DNB considerations) are insensitive to minor changes to steam generator inventory.

The WCAP-12966 (1991) LOCA Analysis concluded that the limiting case was minimum SI and Cd = 0.4 which results in a PCT of 2149 F. The 30 % analysis confirms in section 6.6 that this is the limiting break and calculates a new revised PCT. Based on the WCAP-13707 analysis and a new large break LOCA P-bar-HA sensitivity study, the new PCT is 2122 F.

There are primarily two aspects of the ECCS LOCA Analysis which should be addressed as a result of the reduction in TDF: (a) consideration of the RCS flow; and, (b) consideration of effects of RCS temperature distribution. Within reasonable limits, such as the reduction being proposed, RCS flow has a generally insignificant effect because the break flow dominates the transient almost immediately for both small break (SBLOCA) and large break (LBLOCA). Therefore, the majority of the effect is realized through any changes to RCS Tavg which result. The initial RCS temperature assumed by the LOCA analyses is based upon 100% power design conditions. Applying this methodology to the TDF reduction, no change to RCS Tavg is predicted, and no PCT penalty or benefit is incurred, and the 10CFR50.46 PCT limit of 2200 F is maintained.

The following are not included in the WCAP: (1) The Post-LOCA Containment mass and energy release. The reduction in primary side volume due to SGTP will reduce primary system mass slightly. The impact of reducing core flow was evaluated. Programmed Tavg remains the same even though the temperature difference goes up, and the average volumetric temperature goes up slightly. This is offset by the decrease in primary mass due to the reduction in tube volume. The combined effect is that the current UFSAR values remain bounding. (2) Post LOCA Hot Leg Switchover time: Hot leg switchover (HLSO) time is dependent upon power level and upon RCS, RWST, and accumulator water volumes, and on boron concentrations. The TDF reduction has no effect on the listed parameters, and the effect of the increased SGTP on the RCS initial volume is insignificant, therefore post-LOCA HLSO time is not affected.(3)The Balance -of -Plant (BOP) interface has not significantly changed from the evaluation performed for the 20% plugging limit, and does not effect any analysis. (Reference Part 1 item A.3). (4)Reactor Coolant Loop Stress Reconciliation: The impact of the TDF reduction was evaluated. Based on the slight increase in hot leg temperature, and the reduction in cold leg temperature, it has been determined that the change in system parameters will have an insignificant effect on the design margin.

Since safety design requirements continue to be met and the integrity of the reactor coolant system pressure boundary is not challenged, the assumptions employed in the calculation of the offsite radiological doses remain valid.

A.6. Identify the design basis accidents, if any, for which failure modes associated with the change can be an initiating event.

Reference Introduction Item 4; there are none.

A.7. Discuss the effect of the change on the probability of occurrence of the design basis accidents identified in Part 1, A.6.

There is no adverse effect as no DBAs were identified.

B. Evaluation Questions

B.1. Based on Part 1, A.2, MAY the proposed change increase the probability of occurrence of a malfunction of equipment important to safety evaluated previously in the UFSAR?

YES ____________ NO __X__.

B.2. Based on Part 1, A.3, MAY the proposed change increase the consequences of a malfunction of equipment important to safety evaluated previously in the UFSAR?

YES ____________ NO __X__.

B.3. Based on Part 1, A.5, MAY the proposed change increase the consequences of an accident evaluated previously in the UFSAR?

YES ____________ NO __X__.

B.4. Based on Part 1, A.6 and A.7, MAY the proposed change increase the probability of occurrence of an accident evaluated previously in the UFSAR?

YES ____________ NO __X__

IF THE ANSWER TO ANY OF THE ABOVE QUESTIONS IS YES
THE CHANGE REPRESENTS AN UNREVIEWED SAFETY QUESTION.

PART 2: POTENTIAL FOR CREATION OF A NEW TYPE OF UNANALYSED EVENT

A. Supporting Information

A.1. Based on Part 1, assess the impact of the change and/or failure modes associated with the change to determine if the impact has modified the plant response to the point where it can be considered a new type of accident. Discuss the basis for this determination.

Based on Part 1, all aspects of the proposed new analysis have been evaluated and no new or different type of accident or failure mode has been identified.

A.2. Determine if the failure modes of equipment important to safety associated with the change represent a new unanalyzed type of malfunction. Discuss the basis for this determination.

It does not. The increased tube plugging does not challenge the integrity of the steam generators and does not change any safety-related equipment or affect its ability to perform its intended safety function.

B. Evaluation Questions

B.1. Based on Part 2, A.1, MAY the proposed activity create the possibility of an accident of a different type than any evaluated previously in the UFSAR?

YES __________ NO __X__.

B.2. Based on Part 2, A.2, MAY the proposed activity create the possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the UFSAR?

YES __________ NO __X__.

IF THE ANSWER TO ANY OF THE ABOVE QUESTIONS IS YES,
THE CHANGE REPRESENTS AN UNREVIEWED SAFETY QUESTION.

PART 3: IMPACT ON THE MARGIN OF SAFETY

A. Supporting Information

A.1. Identify the acceptance limits which form the licensing basis for the TECHNICAL SPECIFICATIONS (i.e., the accident analysis and other design basis) that could be affected by the change.

A Technical Specification change request has been DLC approved and submitted to NRC. Pending acceptance of this proposed technical specification change, no other technical specifications are revised by the proposed change. The following acceptance limits were identified within the WCAP: DNBR greater than the safety analysis limit value; PCT < 2200F (LOCA) and < 2700F (Non-LOCA);Containment Pressure less than Design; Peak RCS pressure is less than that which would cause stresses to exceed the Faulted Condition Stress Limits.

A.2. Discuss the impact of the change on the acceptance limits which form the basis for the TECHNICAL SPECIFICATIONS.

As described in A.1, there is no additional change to the technical specifications, no change to the acceptance limits, and no impact on the basis. All LOCA and Non-LOCA accidents have been evaluated or re-analyzed for the effects of the increased tube plugging. All acceptance criteria continue to be met.

B. Evaluation Question

B.1 Based on Part 3, A.1 and A.2, Does the proposed activity reduce the margin of safety as defined in the basis for any TECHNICAL SPECIFICATION?

YES ________ NO ___X___.

IF THE ANSWER TO ANY OF THE ABOVE QUESTIONS IS YES, THE CHANGE REPRESENTS AN UNREVIEWED SAFETY QUESTION.

PART 4: 10CFR50.59 EVALUATION CONCLUSION

Based on the evaluation in Parts 1, 2, and 3 the change:

_X__ Does not involve an UNREVIEWED SAFETY QUESTION.

____ Involves an UNREVIEWED SAFETY QUESTION. Contact Nuclear Safety Department before presenting to the Onsite Safety Committee.

PART 5: ENVIRONMENTAL EVALUATION (applicable to BVPS Unit 2)

A. Supporting Information

A.1. Identify any significant increase in any adverse environmental impact previously evaluated in the Final Environmental Statement - Operating License stage, environmental impact appraisals, or in any decisions of the Atomic Safety and Licensing Board.

N/A: The proposed change is for Unit 1

A.2. Identify any significant change in effluents or power level.

There are none.

A.3. Identify any matters, not previously reviewed and evaluated in the Environmental Protection Plan, Final Environmental Statement - Operating License Stage, or NPDES permit, which MAY have a significant adverse environmental impact.

There are none.

B. Evaluation Question

B.1. Based upon Part 5, A.1, A.2 and A.3, the change:

_X_ Does not involve an UNREVIEWED ENVIRONMENTAL QUESTION.

____ Involves an UNREVIEWED ENVIRONMENTAL QUESTION. Contact Nuclear Safety Department before presenting to the Onsite Safety Committee.

PART 6: PREPARATION, REVIEW AND APPROVAL DOCUMENTATION

Ronald C. Wittschen — 8-5-93
Preparer's Signature — Date

[illegible signature] — 8/5/93
Independent Reviewer's Signature — Date

[illegible signature] — 8/11/93
(other Approval Signature) — Date

OSC Concurrence: Meeting Number BV____-OSC-_32_-_93_.
8/12/93