APR 27 1979

Docket Nos. 50-338
and 50-339

Mr. W. L. Proffitt
Senior Vice President - Power
Operations
Virginia Electric & Power Company
P. O. Box 26666
Richmond, Virginia 23261

Dear Mr. Proffitt:

SUBJECT: ISSUANCE OF AMENDMENT NO. 10 TO FACILITY OPERATING LICENSE NO. NPF-4 - NORTH ANNA POWER STATION, UNIT NO. 1

The Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 10 to Facility Operating License NPF-4.

This Amendment makes changes to Appendix A Technical Specifications which involve:

1. inservice inspection of flow splitter plates;

2. displacement of reactor coolant pumps, and

3. loose parts monitoring

Our Safety Evaluation on this matter is enclosed. As discussed with you we have determined and you have agreed that these technical specification changes are required.

We also agreed upon this action as a Class III amendment pursuant to 10 CFR Part 170 which requires a fee of $4,000.00. Please provide us with the remittance fee for this amendment as soon as possible.

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OFFICE➤						
SURNAME➤						
DATE➤						

APR 27 1979

Also enclosed is a copy of the notice concerning the issuance of Amendment No. 10 which has been forwarded to the Office of the Federal Register for publication.

Sincerely,
Original Signed by
O. D. Parr

Olan D. Parr, Chief
Light Water Reactors Branch No. 3
Division of Project Management

Enclosures:
1. Amendment No. 10 to NPF-4
2. Safety Evaluation Report
3. Federal Register Notice

ccs w/enclosure:
See next page

OFFICE	DPM:LWR #3	DPM:LWR #3	OELD	DPM:LWR #3		
SURNAME	MRushbrook:ab	ADromerick	DSwanson	ODParr		
DATE	4/27/79	4/27/79	4/27/79	4/27/79		

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-338

NORTH ANNA POWER STATION, UNIT NO. 1

FACILITY OPERATING LICENSE

License No. NPF-4
Amendment No. 10

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

   A. The issuance of this license amendment to Virginia Electric and Power Company complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations as set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the license, as amended, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Appendix A Technical Specifications as indicated in the attachment to this license amendment. Facility Operating License No. NPF-4 is hereby amended to read as follows:

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OFFICE➤						
SURNAME➤						
DATE➤						

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 10, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/S/

Domenic B. Vassallo, Assistant Director
for Light Water Reactors
Division of Project Management

Date of Issuance: APR 27 1979

Enclosure:
Revised pages
to Appendix A Technical Specifications

See previous yellow for concurrences

OFFICE	LWR #3:LA	LWR #3:LPM	OELD	LWR #3:BC	LWR:AD
SURNAME	MRushbrook/LM	ADromerick	DSwanson	ODParr	DBVassallo
DATE	4/27/79	4/27/79	4/ /79	4/27/79	4/27/79

(2) <u>Technical Specifications</u>

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 10, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Olan D. Parr, Chief
Light Water Reactors Branch No. 3
Division of Project Management

Date of Issuance:

Enclosure:
Revised pages
to Appendix A Technical Specifications

concurrence limited to legal review of amendment

OFFICE ➤	LWR #3:LA	LWR #3:LPM	OELD	LWR #3:BC		
SURNAME ➤	MRushbrook/LM	ADromerick	DSwanson DL	ODParr		
DATE ➤	4/27/79	4/27/79	4/27/79	4/ /79		

ATTACHMENT TO LICENSE AMENDMENT NO. 10

FACILITY OPERATING LICENSE NO. NPF-4

DOCKET NO. 50-338

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

Pages

IV
3/4 3-56 (added)
3/4 3-57 (added)
3/4 4-31
3/4 4-31a (added)
B 3/4 3-3

APR 27 1979

NORTH ANNA POWER STATION, UNIT NO. 1

AMENDMENT NO. 10 TO NPF-4

Distribution w/enclosure

Docket File
NRC PDR
Local PDR (2)
LWR #3 File
D. Swanson, OELD
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R. Boyd
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O. Parr
A. Dromerick
M. Rushbrook
M. Rushbrook
F. Williams
A. Toalston, AIG
I. Dinitz, AIG
L. Cobb, I&E
I&E (3)
N. Dube, MPA
M. Jinks (4)
W. Miller, ADM
R. Diggs, ADM
B. Scharf (10)
E. Hughes, DPM
D. Skovholt, DPM
P. Wagner, DOR
A. Schwencer, DOR
C. Parrish, DOR
E. Reeves, DOR
R. DeYoung, DSE
V. Moore, DSE
M. Ernst, DSE
W. Kreger, DSE
R. Denise, DSE
W. Regan, DSE
P. Leech, DSE
M. Duncan, DSE
R. Mattson, DSS
J. Knight, DSS
S. Hanauer, DSS
R. Tedesco, DSS

CP 1

6D

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

SURVEILLANCE REQUIREMENTS (Continued)

a. If the absolute value of $\frac{R_{ij}-\overline{R}_j}{\overline{R}_j}$ is greater than $2\sigma_j$, another map shall be completed to verify the new $\overline{R}_j$. If the second map shows the first to be in error, the first map shall be disregarded. If the second map confirms the new $\overline{R}_j$, four more maps (including rodded configurations allowed by the insertion limits) will be completed so that a new $\overline{R}_j$ and σ_j can be defined from the six new maps.

4.3.3.8.2 The APDMS shall be demonstrated OPERABLE:

a. By performance of a CHANNEL FUNCTIONAL TEST within 7 days prior to its use and at least once per 31 days thereafter when used for monitoring $F_j(Z)$.

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

INSTRUMENTATION

LOOSE PARTS MONITORING SYSTEMS

LIMITING CONDITIONS FOR OPERATION

3.3.3.9 The loose parts monitoring system instrumentation identified in Table 3.3-12 shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION

If all channels of one or more collection regions are inoperable, restore the instrument(s) to OPERABLE status within 30 days or, in lieu of any report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the channels to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.3.3.9 Each channel of the loose parts monitoring system identified in Table 3.3-12 shall be demonstrated OPERABLE by the performance of:

a. A CHANNEL CHECK at least once per 24 hours.

b. A CHANNEL FUNCTIONAL TEST at least once per 31 days.

c. A CHANNEL CALIBRATION at least once per 18 months.

TABLE 3.3-12

LOOSE PARTS MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE
1. Steam Generator Transducers	1/steam generator
2. Reactor Vessel Flange Transducers	1/2
3. Reactor Vessel Lower Plenum Transducers	1/2

REACTOR COOLANT SYSTEM

3/4.4.10 STRUCTURAL INTEGRITY

ASME CODE CLASS 1, 2 & 3 COMPONENTS

LIMITING CONDITION FOR OPERATION

3.4.10.1 The structural integrity of ASME Code Class 1, 2 and 3 components shall be maintained in accordance with Specification 4.4.10.1.

APPLICABILITY: ALL MODES.

ACTION:

a. With the structural integrity of any ASME Code Class 1 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature more than 50°F above the minimum temperature required by NDT considerations.

b. With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 200°F.

c. With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) from service.

d. With any RCP shaft deflection indication greater than 20 mils, the reactor shall be placed in at least HOT STANDBY within 1 hour, the affected RCP(s) tripped and then affected flow straightener plate(s) ultrasonically examined.

e. The provisions of Specification 3.0.4 are not applicable.

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS

4.4.10.1 In addition to the requirements of Specification 4.0.5, 1) the Reactor Coolant pump flywheels shall be inspected per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975 and 2) the flow straighteners in each steam generator-to-RCP elbow shall be ultrasonically examined whenever a RCP shaft deflection of greater than 20 mils is indicated and at least once per 18 months.

REACTOR COOLANT SYSTEM

STRUCTURAL INTEGRITY

STEAM GENERATOR SUPPORTS

LIMITING CONDITION FOR OPERATION

3.4.10.2 The temperature of the steam generator supports shall be maintained:

a. > 225°F for A572 material monitored at a middle level corner during operation and at a top level corner during heatup of the supports.

b. < 355°F at the monitored top level corner.

c. > 85°F for A36 material monitored at a bottom level corner during heatup.

APPLICABILITY: With pressurizer pressure > 1000 psig.

ACTION: With the temperature of any steam generator support outside the above limits, restore the temperature to within the limit within 4 hours or be below 1000 psig within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.10.2.1 The steam generator support temperatures for A572 material shall be verified to be within the specified limits at least once per 12 hours.

4.4.10.2.2 The steam generator support temperatures for A36 material shall be verified to be within the specified limit prior to exceeding a pressurizer pressure of > 1000 psig.

4.4.10.2.3 In addition to the requirements of Specification 4.0.5, at least one third of the main member to main member welds, joining A572 material, in the steam generator supports, shall be visually examined during each 40 month inspection interval.

3/4.3.3.6 POST-ACCIDENT INSTRUMENTATION

The OPERABILITY of the post-accident instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident.

3/4.3.3.7 FIRE DETECTION INSTRUMENTATION

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

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

3/4.3.3.8 AXIAL POWER DISTRIBUTION MONITORING SYSTEM (APDMS)

OPERABILITY of the APDMS ensures that sufficient capability is available for the measurement of the neutron flux spatial distribution within the reactor core. This capability is required to (1) monitor the core flux patterns that are representative of the peak core power density, and (2) limit the core average axial power profile such that the total power peaking factor F_Q is maintained within acceptable limits.

3/4.3.3.9 LOOSE PARTS MONITORING SYSTEM

OPERABILITY of the Loose Parts Monitoring System provides assurance that loose parts within the RCS will be detected. This capability is designed to ensure that loose parts will not collect and create undesirable flow blockages.

SAFETY EVALUATION REGARDING STRUCTURAL INTEGRITY OF THE REACTOR COOLANT PUMP SUCTION ELBOW SPLITTER, NORTH ANNA UNIT 1

Background

On March 21, 1979, we were advised by the Virginia Electric and Power Company (VEPCO) that during a routine cleaning of the reactor coolant loop crossover leg pipes in North Anna Unit 2 cracks were discoverd in splitter plate 2-C which is installed in the reactor coolant system pipe elbow leading into the suction side of the reactor coolant pump. The flow splitter plates are not structural members and were installed in the North Anna Units 1 and 2 reactor coolant system to enhance flow distribution at the pump impeller inlet and to increase uniformity in velocity distribution.

During a telephone conversation with the Office of Inspection and Enforcement (I&E) on April 5, 1979, VEPCO agreed that North Anna Unit 1 would not be returned to service until technical justification acceptable to the NRC was provided to show that the flow splitter plates installed in North Anna Unit 1 are structurally sound. I&E transmitted a letter dated April 6, 1979 to VEPCO regarding the confirmation of action.

At the request of VEPCO, a meeting was held on April 12, 1979 in Bethesda, Maryland to allow the licensee an opportunity to discuss this matter with the staff. At this meeting, VEPCO presented a fatigue analysis to demonstrate that cracking similar to that observed in Unit 2 would not occur in the flow splitter plates in Unit 1. On the basis of the analysis, VEPCO concluded that the Unit 1 flow splitter plates would not fail because those flow splitter plates had already accumulated sufficient fatigue cycles beyond the material endurance limit to preclude failure during subsequent operation. However, no differences in flow splitter design or flow conditions could be identified by VEPCO for Units 1 and 2. Also presented were the results from ultrasonic examinations, which were performed over the length of the Unit 1 flow splitter plates. The ultrasonic technique was developed using Unit 2 to calibrate the equipment and then applied to Unit 1. The results of the ultrasonic examination indicated that the Unit 1 flow splitter plates were free of the severe cracking found at Unit 2. However, two discontinuities, one three inches long located 22 inches from the leading edge of the flow splitter plate and a second 1/4 inch long two inches from the previous indication, were found during the examination of the Loop B, flow splitter plate. The discontinuities are located at the junction of the lateral and longitudinal welds of the flow splitter plate. VEPCO states that these discontinuities resulted from defects present in the structure from fabrication and differ from the service induced flaws present in the Unit 2 flow splitter plate.

Based on our evaluation of the material, design and flow conditions associated with the flow splitter plates in Units 1 and 2, we concluded that there is little margin against fatigue cracking for the Unit 1 flow splitter plates and thus concluded that VEPCO's fatigue analysis to demonstrate the integrity of the Unit 1 flow splitter plates was unacceptable. Subsequently, we required the licensee to postulate failure of the flow splitter plate to ensure that a failed plate would not lead to unacceptable safety consequences. We required the postulated flow splitter plate analysis to cover a spectrum of sizes ranging from small sizes that would pass through the pump into the core to a huge piece that would cause pump failure. To further ensure structural integrity, we required VEPCO to develop an inservice inspection program to periodically inspect the flow splitter plates and reduce the potential for continued operations with severely cracked flow splitter plates. In response to our requirements expressed at the April 12, 1979 meeting, VEPCO, on April 15, 1979, submitted the requested safety evaluation report and inservice inspection program.

We have reviewed the information submitted by VEPCO and our evaluation of this matter is discussed in the following paragraphs.

Evaluation

The flow splitter plate under consideration is installed in a 31-inch, 90 degree elbow in the reactor coolant pump suction. From a consideration of pump geometry and the fracture characteristics of the failed flow splitter plate, a range of fragment sizes was established by the licensee. The largest piece which could pass through the pump was estimated to be nine inches by nine inches by 1-1/16 inch thick. The smallest fragment considered to be a 1-1/16 inch cube.

Based on our review, we have determined that the postulated flow splitter plate failure sizes adequately represent failure sizes that likely would result should cracking similar to that in Unit 2 occur.

The potential consequences of fragments in the estimated size range are (1) pump damage sufficient to create a reduction in loop flow, (2) damage to reactor vessel internals and instrument tubing should the flow splitter plate fragments manage to impact them and (3) the occurrence of flow blockage at various locations in the reactor vessel.

With respect to the reactor coolant pump, we have determined that if the largest postulated piece of failed flow splitter plate passes through the pump, the most likely result would be impeller key failure with loss of pump flow. Deformation of the impeller and diffuser vanes, most probably without fracture of these parts, would be anticipated. Abnormal pump shaft bending loads are to be expected. However, the shaft deflection resulting from these loads would alert plant operators via the installed reactor coolant pump shaft displacement monitoring system to stop the pumps before shaft failure occurs. Based on the above, we conclude that a technical specification is required which will require the licensee to shut down the plant and nondestructively examine all flow splitter plates whenever a pump shaft proximity probe indicates excessive deflection. We concur with the licensee that the reactor piping including the flow splitter elbow will not fail from the impact of a postulated failed portion of the splitter plate.

In the event that a failed portion of a flow splitter plate should manage to reach the interior of the reactor vessel and impact on the internals structure, there could be some local deformation. However, American Society of Mechanical Engineers Boiler and Pressure Vessel Code level D limits should not be exceeded. Vessel bottom mounted in-core instrumentation tubing will deform plastically if impacted so as to pinch off the tubing and cause instrument function lost. These instruments however are not required for any safety functions if the postulated event were to occur.

On the basis of our review, we have concluded that the consequences of a failed splitter plate causing damage to reactor piping, reactor coolant pump and vessel internals have been evaluated and are considered acceptable for the postulated event. The pressure boundary will not be breached, but loss of reactor coolant pump flow and loss of some in-core instrumentation capability should be expected, together with local deformation within the reactor coolant pump and portions of the reactor vessel internals.

With respect to the consequences of fragments sufficient to damage the reactor coolant pump and create a reduction in loop flow and the occurrence of flow blockage of various locations in the reactor vessel, our evaluation of these situations is as follows. Also considered in the evaluation is the use of the installed loose parts monitoring system in detecting postulated splitter element fragments.

One consideration is the fuel rod behavior effect of coolant flow blockage, either within the coolant channels of a fuel assembly or in the reactor vessel lower internals. The licensee has considered a flow splitter plate piece simultaneously covering four lower core plate flow holes directly below a fuel assembly. Separate consideration was given to a smaller piece entering one of the core plate flow holes. The licensee has referenced the results of analyses provided on the "Standard Reference System Design RESAR-3S, Westinghouse Electric Corporation Safety Analysis Report". For that design it was predicted for complete blockage of the fuel assembly inlet nozzle that full flow recovery would occur about 30 inches downstream of the blockage. For an assumed 41 percent blockage occurring in the axial center of a fuel bundle, flow recovery was predicted to take place four to five inches beyond the blockage. This is illustrated schematically in Figure 1 (attached). Also illustrated on Figure 1 is an anticipated axial power shape at beginning of life (1.55 cosine skewed toward the bottom of the core by a -20 percent axial offset). Under these conditions the areas of reduced flow occur in a region of less than maximum peaking factor.

Based on the above analysis, we conclude that the worst condition would be occurrence of departure from nucleate boiling on a maximum of four rods caused by a particle lodged in a single channel. Blockage at the entrance is not expected to cause departure from nucleate boiling in the affected fuel assembly.

Since the lower end fitting flow holes in the North Anna design are not of sufficient size to permit fragments greater than one-half inch to enter the fuel assembly, a flow blockage in the fuel assembly greater than that caused by the local reduction between fuel rods is not considered possible. This is conditioned by a further definition of the smallest particle size. However, a small particle assumed to lodge in a fuel assembly would create a blockage less than that already analyzed, or would create turbulence. We conclude that the flow blockage resulting from the failure of the reactor coolant pump flow splitter plate will not result in a departure from nucleate boiling condition more extreme than departure from nucleate boiling on four rods for each fragment lodged in a fuel assembly.

The licensee has provided a discussion of the effects of a possible degradation of loop flow caused by the presence of a fragment in the reactor coolant pump. The assumption of a pump impeller key failure with a loss of pump flow would correspond to a partial loss of flow transient. This transient has been previously analyzed by the licensee in the North

Anna Unit 1 Final Safety Analysis Report with the conclusion that a minimum departure from nucleate boiling ratio of 1.3 is not exceeded. The analysis of the partial loss of flow transient provided in the Final Safety Analysis Report predicted reactor trip at 87 percent loop flow and 97 percent core flow. The analysis included a measurement uncertainty in flow measurement. A reactor trip signal from the pump breaker position is provided as an anticipatory signal which serves as a backup to the low flow signal. A flow degradation less than the complete loss of one pump may not generate a trip on low flow. However, the reduction in departure from nucleate boiling ratio due to flow reduction to the 87 percent (trip) level is less severe.

We have considered the effect of partial loss of flow combined with flow blockage. Since inlet flow blockage does not appreciably affect the bundle flow at elevations higher than 30 inches in the core, we expect no significant effect on the minimum departure from nucleate boiling previously calculated for the transient. Our conclusions regarding reduction in departure from nucleate boiling for the combination of events remains unchanged.

Based on our evaluation of the licensee's submittal and pertinent analyses in the North Anna Unit 1 Final Safety Analysis Report, we conclude that the effects of flow blockage resulting from failure of the reactor coolant pump elbow splitter would result in the occurrence of departure from nucleate boiling on a maximum of four rods for each fragment lodged in a fuel assembly. Further, the combined effects of flow blockage and flow degradation due to pump damage would result in no significant additional effects on previously analyzed transients. We have concluded that the limited fuel damage which might result from the postulated events is acceptable.

We have also concluded that the combined effects of flow blockage due to loose parts and the occurrence of those accidents evaluated in Chapter 15 of the Final Safety Analysis Report (low probability events) need not be considered provided that prompt detection of the loose parts and corrective action is taken.

The prompt detection of loose parts will be accomplished by the loose parts monitoring system provided for North Anna Unit 1. This system was evaluated against the guidelines of Regulatory Guide 1.133 (out for comment), "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors." The present system meets the minimum requirements of two sensors located at each natural collection region and has the minimum sensitivity suggested by the guide.

On the basis of our review, we are incorporating in the North Anna Unit 1 Technical Specifications the following requirements:

(a) The location of the sensors.

(b) A limiting condition for operation requiring the loose-parts detection system to be operable during startup and power operation. If all channels of one or more collection regions are inoperable for more than 30 days, the reactor need not be shutdown, but a special report should be prepared and submitted to the Commission within the next 10 days outlining the cause of the malfunction and the plans for restoring the channel(s) to an operable status.

(c) A surveillance requirement that each channel of the loose-parts detection system be demonstrated operable by a channel check performed at least once per 24 hours, a channel functional test performed at least once per 31 days, and a calibration test performed at least once per 18 months.

With respect to the inservice inspection program for the flow splitter plates, we have determined on the basis of our review that the licensee's proposed inservice inspection program for the flow splitter plates is unacceptable. Therefore, we are incorporating in the North Anna Unit 1 Technical Specifications an inservice inspection program consisting of the following:

(1) Ultrasonic examinations of the elbows containing the flow splitter plates shall be conducted at each refueling outage for all loops.

(2) The ultrasonic examinations shall be performed using the method developed and calibrated using the large flaws in North Anna Unit 2. The details of the examination are contained in Procedure 1-TP-1, "Ultrasonic Test Procedure for Examining Splitter Plates in Steam Generator-to-Pump Elbows North Anna Power Station Unit 1." The applicability of the procedure was demonstrated to IE Regional Office II personnel, who witnessed the testing and concluded the procedure is adequate to detect large cracks similar to those in North Anna Unit 2.

(3) Reports of the examination results will be submitted to the NRC for review. The report should contain a determination regarding the growth of any existing flaws in the structure and identification of any new flaws that might have occurred in the interim service period.

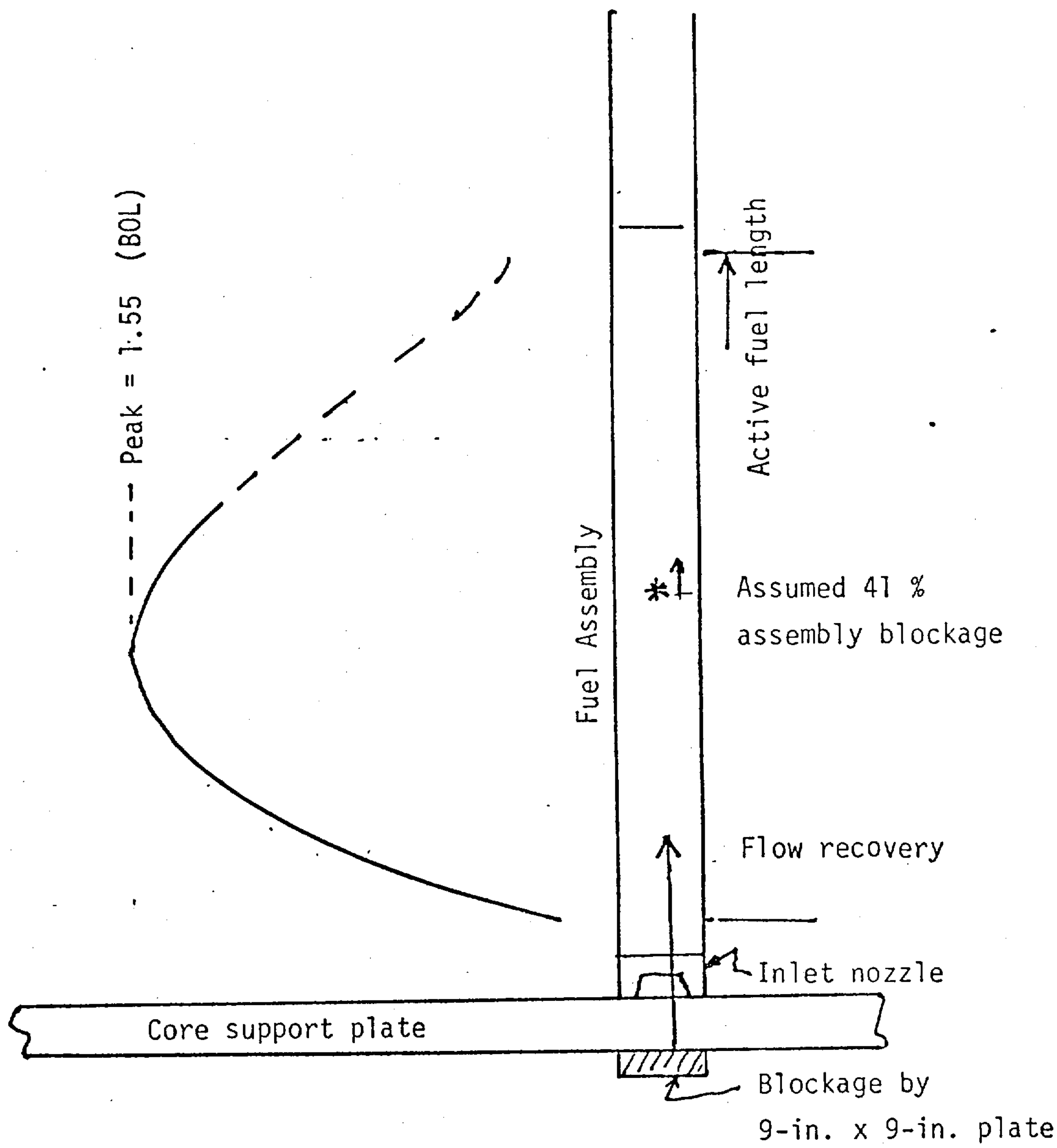
Environmental Consideration

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR Section 50.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered or a significant decrease in any safety margin, it does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public. Also, we reaffirm our conclusions as otherwise stated in our Safety Evaluation Report and its Supplements.

Figure 1


Peak = 1.55 (BOL)
Active fuel length
Fuel Assembly
Assumed 41 %
assembly blockage
Flow recovery
Inlet nozzle
Core support plate
Blockage by
9-in. x 9-in. plate

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-338

VIRGINIA ELECTRIC AND POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment 10 to the Facility Operating License No. NPF-4, issued to Virginia Electric and Power Company, which requires an inservice inspection program to periodically inspect the flow splitter plates installed in the reactor coolant system pipe elbow adjacent to the reactor coolant pump. Changes have been made to the Appendix A Technical Specifications regarding the inservice inspection of flow splitter plates, the displacement monitoring system, and a limiting condition for operation requiring the loose-parts detection system to be operable during startup and power operation. Amendment No. 10 is effective as of its date of issuance.

The Amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration. 7906070325

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR Section 51.5(d)(4) an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

OFFICE➤						
SURNAME➤						
DATE➤						

For further details with respect to this action see a copy of (1) Amendment No. 10 to NPF-4, and (2) the Safety Evaluation dated April, 1979. These items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. 20555 and at the Board of Supervisor's Office, Louisa County Courthouse, Louisa, Virginia 23093 and at the Alderman Library, Manunscripts Department, University of Virginia, Charlottesville, Virginia 22901. A copy of these items may be obtained upon request to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Project Management.

Dated at Bethesda, Maryland this 27th day of April, 1979.

FOR THE NUCLEAR REGULATORY COMMISSION

/S/

Olan D. Parr, Chief
Light Water Reactors Branch No. 3
Division of Project Management

concurrence limited to legal review of Fed. Reg. notice

OFFICE	LWR #3:LA	LWR #3:LPM	OELD	LWR #3:BC	
SURNAME	MRushbrook/LM	ADromerick	DSwanson	ODParr	
DATE	4/27/79	4/[illegible]/79	4/27/79	4/27/79	