VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

June 22, 2000

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Serial No.	00-307
NL&OS/GSS/ETS	R0
Docket Nos.	50-338
	50-339
License Nos.	NPF-4
	NPF-7

Gentlemen:

VIRGINIA ELECTRIC AND POWER COMPANY
NORTH ANNA POWER STATION UNITS 1 AND 2
PROPOSED TECHNICAL SPECIFICATION CHANGES
ELIMINATION OF SEISMIC EFFECTS FROM
CONTROL ROD DROP TIMES

Pursuant to 10 CFR 50.90, Virginia Electric and Power Company requests amendments to the Facility Operating Licenses NPF-4 and NPF-7 for North Anna Power Station Units 1 and 2, respectively. The proposed changes will add a risk-informed License Condition. The License Condition will permit the elimination of the assumed increase in the rod control cluster assembly drop time resulting from a concurrent trip and seismic event when determining if the measured rod drop times meet the Technical Specifications limit of 2.7 seconds.

Attachment 1, Discussion of Changes, provides the technical basis for our request and is consistent with the NRC's August 16, 1995 Policy Statement on probabilistic risk analysis. This justification is also consistent with the principle elements of risk-informed, plant-specific decisionmaking set forth in Regulatory Guide 1.177. The proposed License Condition and Technical Specifications Basis changes are provided as a mark-up in Attachment 2 and a typed version in Attachment 3.

We have evaluated the changes and have determined that the proposed Technical Specifications changes will not create a significant hazards consideration as defined in 10 CFR 50.92. The basis for our determination that the change does not involve a significant hazards consideration is provided in Attachment 4. We have also determined that operation with the proposed changes will not result in any significant increases in the amounts of effluents that may be released offsite and in any significant increases in individual or cumulative occupational radiation exposure. Therefore, the proposed amendment is eligible for categorical exclusion as set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment is needed in connection with the approval of the proposed changes.

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Virginia Electric and Power Company requests approval of the proposed License Condition by February 2001 with an implementation schedule of 30 days from approval of the License Condition.

If you have any further questions or require additional information, please contact us.

Very truly yours,

David A. Christian
Senior Vice President and
Chief Nuclear Officer

Attachments:

Attachment 1	Discussion of Changes
Attachment 2	Mark-up of Technical Specifications Changes
Attachment 3	Proposed Technical Specifications Changes
Attachment 4	Significant Hazards Consideration Determination

Commitments made in this letter:

1. Measured rod drop time data will be trended to ensure that adverse trends in rod drop time performance are readily identified and evaluated for corrective action.

cc: U.S. Nuclear Regulatory Commission
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COMMONWEALTH OF VIRGINIA)
)
COUNTY OF HENRICO)

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by David A. Christian who is Senior Vice President and Chief Nuclear Officer of Virginia Electric and Power Company. He has affirmed before me that he is duly authorized to execute and file the foregoing document in behalf of that Company, and that the statements in the document are true to the best of his knowledge and belief.

Acknowledged before me this 22nd day of June, 2000.

My Commission Expires: 3/31/04.

Maggie McClure
Notary Public

(SEAL)

Attachment 1

Discussion of Changes

North Anna Power Station
Units 1 and 2
Virginia Electric and Power Company

DISCUSSION OF CHANGE

Introduction

Virginia Electric and Power Company (Virginia Power) is currently considering use of an advanced fuel product in North Anna cores, which incorporates a number of features to improve fuel reliability, ease of use, and increase DNBR margins for anticipated transients. However, the fuel product design also may cause a modest increase in measured rod control cluster assembly (RCCA) drop times. The expected actual drop times will be within the 2.7 second limit in TS LCO 3.1.3.4 and the accident analyses. However, with the new fuel product, it is anticipated that when the uncertainties, including the seismic allowance, are considered the 2.7 second rod drop limit may not be met for all RCCAs.

One possible option would be to re-perform the accident analyses and increase the 2.7 second analysis limit on the drop times to increase the available margin to account for the potential increase in rod drop time for the new fuel design. Based on our review of this option, we believe that such a change would also necessarily involve concurrent protection system changes (such as reductions in the high pressurizer pressure and/or low RCS flow reactor trip setpoints). Such changes would have the potential to reduce normal operating margin and increase the potential for reactor trip events and associated plant and equipment transients.

Therefore as a result of these considerations, Virginia Power is proposing a risk informed license condition that permits elimination of the assumed increase in the RCCA drop time resulting from a concurrent trip and seismic event when determining if the measured rod drop times, including measurement uncertainties, meet the Technical Specification LCO and safety analysis limit of 2.7 seconds.

The proposed change does not alter the limiting results of the safety analyses presented in Chapter 15 of the UFSAR and safety margins are maintained. The proposed change will not alter the ability of the reactor protection and control system to perform their design functions or to meet the applicable criteria set forth in the IEEE and ANSI standards and in 10 CFR 50 Appendix A. Analyses continue to demonstrate that the RCCAs will perform their design function by inserting into the core following a reactor trip, even with a postulated concurrent seismic event. However, a postulated seismic event concurrent with an event leading to reactor trip could lead to control rod insertion times slightly in excess of those assumed in the accident analysis. The proposed change results in an insignificant increase in core damage frequency or risk. However, the proposed change is conservatively judged to be a malfunction of a type different than already analyzed and therefore, an unreviewed safety question.

Background

On September 6, 1990, the USNRC issued Amendments No. 139 and 122, for facility operating licenses for North Anna Units 1 and 2, respectively. These amendments increased the control rod drop time requirements specified in LCO 3.1.3.4 from a previous value of 2.2 seconds to a new value of 2.7 seconds. This change to the allowable control rod drop time was requested to support a planned fuel design change

from the Westinghouse Low Parasitic 17 x 17 (LOPAR) fuel assembly to a new 17 x 17 assembly with Westinghouse VANTAGE 5H fuel assembly design features. The new design, designated North Anna Improved Fuel (NAIF), utilizes Zircaloy grids and smaller diameter thimble tubes. The amendment request was made, and granted, on the basis that the increased drop time was required because of a reduction in thimble tube diameter associated with NAIF.

Discussions with Westinghouse during this same time frame revealed that the recommended increase in drop time from Westinghouse included an allowance to incorporate the effects of a seismic event on the calculated drop time, and was not solely based on the change in the fuel design. Since the 2.7 second drop time specified in LCO 3.1.3.4 was assumed in the accident analyses supporting the LOPAR/NAIF transition, the question arose at the time of NAIF implementation as to whether or not the seismic effect should be taken into account by reducing the rod drop time surveillance test acceptance criterion to account for the seismic allowance. Obviously, the seismic effect cannot be simulated or measured during plant surveillance testing.

Virginia Power concluded that the surveillance test acceptance criteria should be changed to accommodate the seismic allowance. This approach represents our current practice.

Technical Basis and Discussion of Change

The following sections provide our technical justification of the proposed request, and in particular our basis for concluding that our request is consistent with the NRC's policy statement on probabilistic risk analysis (PRA) (Reference 1). This justification is presented in a format which is consistent with the Principle Elements of Risk-Informed, Plant-Specific Decision making set forth in Regulatory Guide 1.177 (Reference 2). That is to say, the technical justification demonstrates that:

1. The proposed change meets the current regulations. No exemption or rule change is being requested.

2. The proposed change is consistent with the defense-in-depth philosophy. Traditional engineering considerations have been used to demonstrate this consistency.

3. The proposed change maintains sufficient safety margins. Traditional engineering considerations have been used to demonstrate that this is the case.

4. The proposed change produces a negligible change in core damage frequency or risk and is consistent with the Commission's Safety Goal Policy Statement.

5. There is no impact on Virginia Power's Configuration Risk Management Program. RCCA drop time performance will continue to be monitored in a manner that ensures a high degree of reliability for the RCCAs to insert upon demand.

Design and Licensing Basis

The Control Rod Drive Mechanisms (CRDMs) and RCCAs are designed to continue to function after a seismic event. This is consistent with General Design Criterion (GDC) 2 (References 4-6):

> *Criterion 2—Design bases for protection against natural phenomena. Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these structures, systems, and components shall reflect: (1) Appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena and (3) the importance of the safety functions to be performed.*

Westinghouse has traditionally taken a conservative approach in applying GDC-2 to the functionality of the RCCAs. Not only were the RCCAs to be confirmed to insert during and following a seismic event, but the RCCA drop time was also calculated to result from a concurrent reactor trip and seismic event (Reference 6).

Since the seismic effect cannot be measured or simulated in surveillance testing, Virginia Power opted to reflect the seismic effect by maintaining the drop time in the Limiting Condition for Operation (LCO) consistent with the safety analysis assumption and reducing the test acceptance criterion for as-found RCCA drop times by a seismic allowance.

This proposed change to the non-LOCA accident analysis basis will not eliminate the requirement for the confirmation of the capability of the RCCAs to insert during and following a seismic event. The only change proposed is the elimination of the assumed increase in the RCCA drop time resulting from a concurrent trip and seismic event when determining if the measured rod drop times, including measurement uncertainties, meet the Technical Specification LCO and safety analysis limit of 2.7 seconds. The technical arguments which demonstrate that this change is consistent with the considerations of GDC-2 are set forth in subsequent discussions. The control rod drop time assumed in the accident analysis remains unchanged. {Note: the large break LOCA analysis does not credit RCCA insertion. The small break LOCA analysis will continue to reflect the effects of a concurrent seismic event in the assumed RCCA drop times, i.e. the analysis value is adjusted upward from the LCO limit to account for the seismic effect}.

Safety Assessment

The proposed change is consistent with the defense-in-depth philosophy. Historical analytical experience has shown that only a few accidents in the UFSAR are impacted, in terms of available margins to acceptance limits related to the integrity of fission

product barriers, by increased control rod drop times. Specifically, these events are:

- Uncontrolled Rod Cluster Control Assembly Bank Withdrawal From A Subcritical Condition (RCS overpressure case) – UFSAR Chapter 15.2.1
- Loss of external electrical load (RCS and main steam system overpressure case)- UFSAR Chapter 15.2.7
- Locked Reactor Coolant Pump Rotor (RCS overpressure and fuel integrity cases)– UFSAR Chapter 15.4.4
- Complete Loss of Forced Reactor Coolant Flow (fuel integrity case) – UFSAR Chapter 15.3.4
- Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection) (fuel integrity case)- UFSAR Chapter 15.4.6.

For all of these events, the following significant conservatism applies:

- The surveillance test ensures that the slowest measured rod has a drop time which is within the acceptance value. Since there is a distribution of drop times and many rods will have a drop time which is significantly less than the acceptance value, this ensures additional conservatism in the analysis.

Overpressure Considerations –Defense in Depth

The proposed elimination of the seismic component from the RCCA drop time allowance in the safety analysis will not adversely affect Virginia Power's ability to demonstrate acceptable margins to the RCS overpressure limit (i.e. transient pressure <110% of design pressure of 2500 psia) in the analyses of certain UFSAR Chapter 15 events which show some sensitivity to the RCCA drop time. The specific events of concern are:

- Uncontrolled Rod Cluster Control Assembly Bank Withdrawal From A Subcritical Condition (overpressure case) –UFSAR Chapter 15.2.1
- Loss of External Electrical Load (overpressure case)- UFSAR Chapter 15.2.7
- Locked Reactor Coolant Pump Rotor (overpressure case) – UFSAR Chapter 15.4.4

For the uncontrolled rod withdrawal from subcritical (overpressure) analysis, the following conservative assumptions are made. The margins inherent in each of these assumptions are not decreased by the proposed change:

- The maximum reactivity insertion rate analyzed is greater than that resulting from the simultaneous withdrawal of the combination of the two control banks having the maximum combined worth at maximum speed.

- Since the magnitude of the power peak reached during the initial part of the transient for any given rate of reactivity insertion is strongly dependent on the Doppler

coefficient, conservative values (low absolute values) as a function of temperature are used.

- Contribution of the moderator reactivity coefficient is negligible during the initial part of the transient because the heat transfer time between the fuel and moderator is much longer than the neutron flux response time. However, after the initial neutron flux peak, the succeeding rate of power change is affected by the moderator reactivity coefficient. A conservative value is used in the analysis to yield the maximum peak heat flux. This value represents core conditions that are only approached for a small fraction of the total core life, near beginning of cycle (BOC).

- The total trip reactivity for the inserted rods is set at a conservatively low value with respect to typical reload core design results. The most reactive RCCA is assumed to stick in its fully withdrawn position after trip. Trip reactivity vs. rod position is calculated based on a conservative bottom-peaked axial power distribution, which effectively delays the reactivity insertion.

- No credit is taken for the operation of pressurizer sprays or pressurizer PORVs.

- The pressurizer safety valves are assumed to relieve at the upper end of the tolerance band on their set pressure as specified in LCO 3.4.3.

For the loss of external electrical load event, the following conservative assumptions in the analysis of the overpressure case ensure defense in depth. The margins inherent in each of these assumptions are not decreased by the proposed change:

- No credit is taken for a direct reactor trip on turbine trip. Rather, reactor trip on high pressurizer pressure is assumed.

- No credit is taken for the steam relieving capacity of the condenser steam dumps for cases where condenser vacuum is available. The most likely scenario leading to loss of condenser vacuum would be a loss of circulating water due to loss of offsite power (LOOP). A LOOP would generate a reactor trip on loss of power to the station service buses, which is not credited in the analysis.

- No credit is taken for the steam relieving capacity of the steam generator atmospheric relief valves.

- The main steam safety valves are assumed to relieve at the upper end of the tolerance band on their set pressure as specified in LCO 3.7.1.

- No credit is taken for pressurizer sprays.

- No credit is taken for pressurizer power operated relief valves.

- No credit is taken for the reactor (Tavg) control system. Control rods are assumed to be in manual control mode.

- The pressurizer safety valves are assumed to relieve at the upper end of the tolerance band on their set pressure as specified in LCO 3.4.3.

- The moderator temperature coefficient of reactivity is assumed to be at a conservative BOC value. This represents a conservative set of core reactivity conditions which is only approached at the very beginning of core life.

- A maximum trip setpoint and time delay for the high pressurizer pressure trip is assumed which conservatively bounds the channel uncertainty and response time.

- Conservatisms pertaining to core physics assumptions include:

    - a bounding low trip reactivity (% delta k/k), calculated by assuming the most reactive RCCA fails to insert into the core
    - a trip reactivity versus RCCA position which is conservatively low in the core; this delays the post-trip power decrease.

It should also be noted that the loss of load overpressure event does not create a challenge to either fuel integrity (DNBR margin remains) or containment integrity (steam releases from the pressurizer would be predominately quenched by the pressurizer relief tank and would not pose a challenge to containment integrity). Elimination of the seismic allowance from rod drop surveillance testing limits will not alter this conclusion. Therefore the independence of physical fission product barriers is not compromised by the proposed change.

For the locked reactor coolant pump rotor event (overpressure case), the following conservative assumptions are made. The margins inherent in each of these assumptions are not decreased by the proposed change:

- The main steam and RCS overpressurization analyses employ a deterministic treatment of key analysis uncertainties (i.e., on power; RCS Tavg, pressurizer pressure; and RCS thermal design flow).

- The analysis assumes a moderator temperature coefficient which is representative of conditions which are only achievable for a few Effective Full Power Hours (EFPH) at the very beginning of cycle.

- No credit is taken for automatic rod control.

- The pressurizer sprays and PORVs are conservatively assumed not to actuate.

- In the main steam and RCS overpressurization analyses, the pressurizer heaters are conservatively assumed to be enabled.

- Although there is no mechanism for early feedwater isolation during a locked

rotor event, the main steam and RCS overpressurization analyses have conservatively assumed initiation of feedwater isolation upon reactor trip.

- The Reactor Protection System (RPS) setpoints assumed in the safety analysis (i.e., the low coolant loop flow rate reactor trip) are demonstrated to be conservative by the inclusion of appropriate uncertainties for process measurement and signal delay.

Fuel Integrity Considerations –Defense in Depth

The proposed elimination of the seismic component from the RCCA drop time allowance in the safety analysis will not compromise Virginia Power's ability to demonstrate acceptable margins to the RCS fuel integrity limits (i.e. transient DNBR > the 95/95 confidence limit for ANS Condition II events, limited fuel failure for ANS Condition III and IV events) in the analyses of certain UFSAR Chapter 15 events which show some sensitivity to the RCCA drop time. The specific events of concern are:

- Complete Loss of Forced Reactor Coolant Flow (fuel integrity case) – UFSAR Chapter 15.3.4
- Locked Reactor Coolant Pump Rotor (fuel integrity/DNB case)– UFSAR Chapter 15.4.4
- Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection) (fuel integrity case)- UFSAR Chapter 15.4.6.

For the Complete Loss of Forced Reactor Coolant Flow (fuel integrity case), the following conservative assumptions are made. The margins inherent in each of these assumptions is not decreased by the proposed change:

- A least negative Doppler Temperature Coefficient and most positive Moderator Temperature Coefficient (MTC) were assumed since these result in higher heat flux at the time of minimum DNBR. Note that this (MTC) condition does not occur at hot full power due to Technical Specifications limits, and is only approached at reduced power for a few EFPH at the beginning of the fuel cycle.

- No credit was taken for direct reactor trip on either the RCP motor bus undervoltage or underfrequency trip. Rather, it was assumed that reactor trip does not occur until the low RCS loop flow setpoint is reached.

- A minimum low flow trip setpoint and maximum trip time delay are assumed which conservatively bound the effects of channel uncertainty and response time.

- It is also assumed that, upon reactor trip, the most reactive control rod assembly is stuck in its fully withdrawn position, resulting in a minimum insertion of negative reactivity.

- An initial radial power peaking factor ($F\Delta H$) corresponding to the Technical

Specification maximum allowable value is assumed.

Both the underfrequency and the undervoltage trip events are analyzed. The minimum DNBRs for the two accidents showed a significant margin to the statistical DNBR limit.

For the locked reactor coolant pump rotor (DNB case) event, the defense in depth considerations are similar to those discussed previously for the overpressure case. Specifically:

- The total trip reactivity for the inserted rods is set at a conservatively low value with respect to typical reload core design results. The most reactive RCCA is assumed to stick in its fully withdrawn position after trip. Trip reactivity versus rod position is calculated based on a conservative bottom-peaked axial power distribution, which effectively delays the reactivity insertion.

- The analysis supports a moderator temperature coefficient which corresponds to conditions which are only achievable for a few Effective Full Power Hours (EFPH) at the very beginning of cycle.

- No credit was taken for automatic rod control.

- The Reactor Protection System (RPS) setpoints assumed in the safety analysis (i.e., the low coolant loop flow rate reactor trip) are demonstrated to be conservative by the inclusion of appropriate uncertainties for process measurement and signal delay.

- The transient analysis for DNB is performed to determine the number of fuel pins that experience DNB as a result of a Locked Rotor or Sheared Shaft event. A fuel pin is assumed to fail if the predicted MDNBR is less than the statistical DNBR design limit. The Locked Rotor DNB event scenario is therefore designed to produce the most limiting DNB response. From an analytical perspective, this goal is achieved by choosing initial conditions and analysis assumptions that will maximize coolant temperature and the power-to-flow ratio, and minimize pressure during the event.

- The analysis demonstrates that the fraction of fuel failure for this event is less than that which has been demonstrated to provide acceptable dose consequences.

    - Two cases were analyzed for the locked rotor offsite dose calculation. Case 1 is a bounding case which assumed the affected SG tubes were uncovered from time zero.

    - Case 2 represents a more realistic condition for which it was assumed no uncovery of the region of primary to secondary leakage occurred in the affected SG during the first 15 minutes of the event. After this time, the stuck SG PORV is assumed shut, stopping releases from the

affected SG. The offsite dose calculation was performed for the two cases described above. The results are presented in UFSAR Table 15.4-13, where each case assumes cladding failure and gap activity release for 13% of the fuel rods.

These results demonstrate that the exclusion area boundary doses for both cases are a small fraction of the 10 CFR Part 100 guidelines. Case 1 represents a very conservative prediction of doses for the locked rotor event. Case 2, while still conservative, provides a more realistic estimate of these doses. Note that these doses are based on the assumption of 13% fuel failure, which is in itself very conservative for the following reasons.

1. For current core loading patterns, Virginia Power uses our statistical DNBR methodology (Reference 3), which demonstrates that no fuel pins experience DNBR below the limit.
2. Even if a rod were to experience DNBR below the limit for the short periods of time representative of this event, the assumption that this leads directly to clad failure is also very conservative, as considerable test experience has shown.

For the rupture of a control rod drive mechanism housing (rod cluster control assembly ejection) (fuel integrity case) the following conservative assumptions are made. The margins inherent in each of these assumptions is not decreased by the proposed change:

1. The assumption that the event itself occurs is a major conservatism. In particular:

   - Mechanical design and quality control procedures intended to preclude the possibility of a rod cluster control assembly (RCCA) drive mechanism housing failure sufficient to allow a rod cluster control assembly to be rapidly ejected from the core are listed below:

     - Each control rod drive mechanism housing is completely assembled and shop-tested at 4100 psi.
     - The mechanism housings are individually hydrotested as they are attached to the head adapters in the reactor vessel head, and checked during the hydrotest of the completed reactor coolant system.
     - Stress levels in the mechanism are not affected by anticipated system transients at power, or by the thermal movement of the coolant loops. Moments induced by the design-basis earthquake can be accepted within the allowable primary working stress range specified by the ASME Code, Section III, for Class I components.

     - The latch mechanism housing and rod travel housing are each a single length of forged type 304 stainless steel. This material exhibits excellent notch toughness at all temperatures that will be encountered. A significant

margin of strength in the elastic range, together with the large energy absorption capability in the plastic range, gives additional assurance that gross failure of the housing will not occur. The joints between the latch mechanism housing and head adapter, and between the latch mechanism housing and rod travel housing, are threaded joints reinforced by canopy-type rod welds. Administrative regulations require periodic inspections of these (and other) welds.

2. Major conservatisms associated with the analysis are as follows:

   - Average core analysis - The point kinetics model of the RETRAN computer code is used to perform the average core transient analysis. This code includes the simulation of prompt and delayed neutrons (using the six-group model), the thermal kinetics of the fuel and moderator and the balance of the NSSS primary and secondary coolant system. Thermal feedback effects are modeled via temperature dependent reactivity coefficients with a detailed multi-region, transient fuel-clad-coolant heat transfer model. Reactivity insertion from the ejection of the control rod and the subsequent reactor trip are accounted for. Since both the axial and radial dimensions are missing, it is necessary to use very conservative methods (described below) of calculating the ejected-rod worth and hot-channel factor.

   - Ejected-rod worths and hot-channel factors - The values for ejected-rod worths and hot-channel factors are calculated using a synthesis of one-dimensional, two-dimensional and three-dimensional calculations. Standard nuclear design codes are used in the analysis. No credit is taken for the flux-flattening effects of reactivity feedback. The calculation is performed for the maximum allowed bank insertion at a given power level, as determined by the rod insertion limits. Adverse xenon distributions are considered in the calculation. The total transient hot-channel factor, Fqt, is then obtained by combining the axial and radial factors. Appropriate margins are added to the results to allow for calculational uncertainties, including an allowance for nuclear power peaking due to fuel densification. It should also be noted that the ejected rod worths and peaking factors are currently calculated based on the assumption that the initial rod bank position corresponds to control rod insertion limits established in the Core Operating Limits Report (COLR). Since Virginia Power units are base load operation plants and control rods are typically at or near the withdrawn position during power operation, criticality with control rods at the insertion limit is a very low frequency condition.

   - Reactivity feedback weighting factors - The largest temperature rises, and hence the largest reactivity feedbacks, occur in channels where the power is higher than average. Since the weight of a region is dependent on flux, these regions have high weights. This means that the reactivity feedback is larger than that indicated by a simple single-channel analysis. Physics calculations were carried out for temperature changes with a flat temperature distribution, and with a large number of axial and radial temperature distributions.

Reactivity changes are compared and effective weighting factors determined. These weighting factors take the form of multipliers that, when applied to single-channel feedbacks, correct them to effective whole-core feedbacks for the appropriate flux shape. In this analysis, although a point kinetics method is used, only a radial weighting factor is applied. In addition, no weighting is applied to the moderator feedback. This very conservative radial weighting factor is applied to the Doppler reactivity feedback of the fuel as a function of the post-ejection radial power peaking factor to account for the missing spatial effect. This weighting factor has been shown to be conservative compared to three-dimensional analysis.

- Moderator and doppler coefficient - The critical boron concentrations at the BOC and EOC were adjusted in the nuclear code to obtain moderator density coefficient curves that are conservative compared to actual design conditions for the plant. As discussed above, no weighting factor is applied to this coefficient. The Doppler reactivity defect is determined as a function of fuel temperature using a two-dimensional steady-state computer code. The resulting curve is conservative compared to design predictions for this plant.

- Delayed neutron fraction, βeff - The accident is sensitive to βeff if the ejected-rod worth is nearly equal to or greater than βeff, as in zero-power transients. To provide for flexibility in designing future fuel cycles, conservative low estimates of βeff were used in the analysis.

- Trip Reactivity Insertion. Conservative low values of total trip reactivity, including the effect of one stuck rod (i.e., the ejected rod) are assumed. The shutdown reactivity is simulated by a conservative curve of trip reactivity insertion versus time after trip.

<u>The proposed change maintains sufficient safety margins.</u>

Because the proposed change will not alter the limiting results of the safety analyses presented in Chapter 15 of the UFSAR, safety margins are maintained. The proposed change will not alter the ability of the reactor protection and control system to perform their design functions or to meet the applicable criteria set forth in the IEEE standards (References 8-9) and in 10 CFR 50 Appendix A. The reactor coolant system and main steam system will continue to meet applicable ASME code requirements (Reference 7).

Analyses continue to demonstrate that the RCCAs will perform their design function by inserting into the core following a reactor trip, even with a postulated concurrent seismic event. The only change proposed is the elimination of the seismic adjustment from the rod drop time surveillance test limits. The present value for rod drop time assumed in the current safety analyses is not altered by this proposed change.

The proposed change results in an insignificant increase in core damage frequency or risk.

The rod drive system, including the control rods themselves, is not explicitly modeled in the North Anna Probabilistic Safety Assessment (PSA) model. The RCCA drop time seismic component will thus have no effect upon the basic events (i.e. the component failure probabilities) in the PSA model. In general the accident sequence analysis in the PSA model is dependent upon the function of each system, including the reactor trip system. As discussed previously, while a seismic event could potentially delay a reactor trip by a marginal amount, it could not plausibly prevent a reactor trip.

The Virginia Power assessment therefore focused on estimating the impact of delayed reactor trip from a seismic event upon the PSA initiating events. For purposes of the assessment, initiating events were classified in two categories: dependent events, or those which could result from seismic activity (e.g. a loss of external electrical load resulting from seismically induced switchyard damage) and independent events, such as a locked reactor coolant pump rotor. The independent events considered here are not expected to occur as a result of seismic activity, but their consequences could be potentially increased by the seismically induced RCCA drop time.

Based on a review of our current safety analyses, Virginia Power concluded that the limiting event from the standpoint of a challenge to the reactor coolant pressure boundary integrity is the loss of external electrical load. This event was discussed earlier in the Engineering Evaluation. A probability estimate of a seismically induced loss of load leading to RCS failure, loss of coolant and subsequent core damage was made. We conclude that the probability of this event is several orders of magnitude below the North Anna PSA baseline Core Damage Frequency (CDF), and also less than the 5. E-7 incremental conditional core damage probability (ICCDP) threshold that is considered small in NUREG-0800 Section 16.1.

For an examination of a limiting independent event, the locked reactor coolant pump (RCP) rotor event was examined. This event was discussed earlier in the Engineering Evaluation. A probability estimate of a locked RCP rotor event coincident with a seismic event was made. This event was assumed conservatively to lead to a LOCA in every case. Even with this conservative assumption, the core damage probability was below the 5.0 E-7 threshold for classification as "small" and several orders of magnitude below the North Anna PSA baseline Core Damage Frequency (CDF).

Regulatory Guide 1.177 requires a three-tiered approach for evaluating the risk associated with proposed Technical Specifications Allowed Outage Time Changes. Virginia Power has applied Regulatory Guide 1.177 as a framework for developing the basis of the proposed change.

- *Tier 1: PRA Capability and Insights.* As noted above, an incrementally increased RCCA drop time will have a small impact upon the North Anna CDF. The Large Early Release Frequency (LERF) impact was not quantified, but is assessed by similar methods as also being "small".

- *Tier 2: Avoidance of Risk-Significant Plant Configurations.* Plant risk is monitored and controlled by the Configuration Risk Management Program set forth in North Anna Technical Specification 6.8.4.g. This program already works to minimize risk and avoid risk-significant configurations. Its ability to do so will not be affected by the proposed change in treatment of control rod drop times in the accident analyses. The proposed change does not affect or change the risk significance of any component or group of components.

- *Tier 3: Risk Informed Configuration Risk Management.* The risk management program presently in place continuously reviews planned maintenance configurations to ensure that risk is maintained at acceptable low levels. Emergent configurations are evaluated via the same tools. Typically, the corrective actions imposed by Technical Specifications provide appropriate compensatory measures. Additional compensatory measures will be added as a part of the 10 CFR 50.65 a(4) implementation program, if ongoing review finds that such measures are warranted.

In conclusion, the proposed change to the safety analysis treatment of allowable RCCA drop time will have a negligible impact upon the Core Damage Frequency at North Anna and on the Configuration Risk Management Program.

As discussed elsewhere, elimination of the seismic adjustment from RCCA drop time testing criteria will not impact the probability of rod insertion in the event of seismic activity. The RCCAs have been designed to insert during and following a Design Basis Earthquake (DBE). The only impact of a postulated DBE concurrent with an event leading to reactor trip is that the drop time may be extended (on the order of 1 second) from the non-seismic measured value.

The impact of the proposed change should be monitored using performance measurement strategies.

Based on the PSA review in Section 4 above, we conclude that the proposed change to eliminate seismic considerations from RCCA drop time surveillance testing criteria will have no impact on Virginia Power's configuration risk management program for North Anna. No new risk-significant equipment outage configurations will be created by this proposed change.

In eliminating the seismic allowance from the surveillance test criteria, Virginia Power does not propose to set the RCCA drop time criterion to the LCO 3.1.3.4 value, which corresponds to the safety analysis assumption for drop time. Rather, we will develop a limit which accounts appropriately for design and measurement uncertainties. Additionally, measured rod drop time data will be trended to ensure that adverse trends in rod drop time performance are readily identified and evaluated for corrective action.

Specific changes

Include the following Licensing Condition to each Operating License:

G./J. (Unit 1 and 2, respectively) Consideration of the effects of a concurrent seismic event on the Rod Cluster Control Assembly (RCCA) drop time is excluded from the non-LOCA accident analyses.

Concurrent with approval this change, we propose to make the following change to the BASIS for Technical Specification 3.1.3.4:

> **The maximum rod drop time restriction is consistent with the assumed rod drop time used in the accident analyses. Measurement with Tavg greater that or equal to 500 °F and with all reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a reactor trip at operating conditions. The surveillance criteria limits for measured drop times are adjusted downward from the TS limit. This ensures that the accident analysis assumptions regarding the negative reactivity insertion rate of the control rod system as a whole remain bounding ~~for a reactor trip concurrent with a postulated seismic event.~~ The adjusted limits account for the safety analysis limits and uncertainties. Additionally, measured rod drop time data is trended to ensure that adverse trends in drop time performance are readily identified and addressed.**

Safety Significance

Virginia Power is proposing to eliminate the seismic allowance from its control rod (RCCA) drop time surveillance testing criteria. We have determined that this proposed license condition poses an unreviewed safety question in that it eliminates an accident consideration that is currently addressed in the UFSAR accident analyses, i.e. any Chapter 15 accident with the effects of a concurrent seismic occurrence reflected in the RCCA drop time.

While a unreviewed safety questions is posed, we believe that the proposed change is acceptable and should be approved on the basis that

- The proposed change meets the current regulations. No exemption or rule change is being requested.
- The proposed change is consistent with the defense-in-depth philosophy. Traditional engineering considerations have been used to demonstrate this consistency.

- The proposed change maintains sufficient safety margins. Traditional engineering considerations have been used to demonstrate that this is the case.
- The proposed change produces a negligible change in core damage frequency or risk and is consistent with the Commission's Safety Goal Policy Statement
- There is no impact on Virginia Power's Configuration Risk Management Program. RCCA drop time performance will continue to be monitored in a manner that ensures a high degree of reliability for the RCCAs to insert upon demand.

References:

1. USNRC, "Use of Probabilistic Risk Assessment Methods in Nuclear Activities: Final Policy Statement," *Federal Register*, Vol. 60, p. 42622, August 16, 1995.
2. USNRC, Regulatory Guide 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications", August 1998.
3. Virginia Power Topical Report VEP-NE-2-A, "Statistical DNBR Evaluation Methodology", June 1987.
4. North Anna UFSAR Section 4.2.3 – Reactivity Control Systems.
5. North Anna UFSAR Section 3.1.2- Design Basis for Protection Against Natural Phenomena.
6. Westinghouse/WOG/NRC Meeting to Discuss Rod Drop Time, August 30, 1990.
7. ASME Code, Section III, Article NB-3000.
8. IEEE Std 279-1971 (UFSAR Section 7.2.2.2).
9. IEEE Std 344-1971 (UFSAR Section 3.10).
10. North Anna UFSAR, Chapter 15, Accident Analysis.

Attachment 2

Mark-up of Unit 1 and Unit 2 Technical Specifications Changes

North Anna Power Station
Units 1 and 2
Virginia Electric and Power Company

F. The design of the reactor coolant pump and steam generator supports may be revised in accordance with the licensee's submittal dated November 6, 1986 (Serial No. 86-477A).

G. ~~Deleted.~~ Insert 1

H. This license is effective as of the date of issuance and shall expire at midnight on April 1, 2018.

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by:
R. C. DeYoung, for

Roger S. Boyd, Director
Division of Project Management
Office of Nuclear Reactor Regulation

Attachments:

1. Construction Related Items to be completed prior to Initial Criticality
2. Appendices A, B, and C
3. Figure 1

Date of Issuance: APR 1 1978

INSERT 1

Consideration of the effects of a concurrent seismic event on the Rod Cluster Control Assembly (RCCA) drop time is excluded from the non-LOCA accident analyses.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original design criteria are met. Misalignment of a rod requires measurement of peaking factors or a restriction in THERMAL POWER; either of these restrictions provides assurance of fuel rod integrity during continued operation. In addition, those accident analyses affected by a misaligned rod are reevaluated to confirm that the results remain valid during future operation.

Continuous monitoring of rod position with respect to insertion limits and rod deviation provided by the rod insertion limit monitor and rod position deviation monitor, respectively. OPERABILITY of the rod position deviation monitor is verified by a functional test at least once per 7 days and by comparison of the indicated positions versus the respective demand position indicators at least once per 12 hours. If the rod position deviation monitor or the rod insertion limit monitor is inoperable, the frequency of manual comparison of indicated rod (or bank) position is increased to an interval of at least once per 4 hours.

In the event that a malfunction of the Rod Control System renders control rods immovable, provision is made for continued operation provided:

- the affected control rods remain trippable, and
- the individual control rod alignment limits are met.

In the event that a malfunction of the Rod Control System renders control rod banks immovable during surveillance testing, provision is made for 72 hours of continued operation provided:

- the affected control rod banks remain trippable,
- the individual control rod alignment limits are met,
- a maximum of one control or shutdown bank is inserted no more than 18 steps below the insertion limit, and
- the SHUTDOWN MARGIN requirements are verified every 12 hours during the period the insertion limit is not met.

Control Bank D is excluded from the 72 hour provision since insertion of D Bank below the insertion limit is not required for control rod assembly surveillance testing.

Checks are performed for each reload core to ensure that bank insertions of up to 18 steps will not result in power distributions which violate the DNB criterion for ANS Condition II transients (moderate frequency transients analyzed in Section 15.2 of the UFSAR) or in a violation of the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 during the repair period.

The 72 hour period for a control rod assembly bank to be inserted below its insertion limit restricts the likelihood of a more severe (i.e., ANS Condition III or IV) accident or transient condition occurring concurrently with the insertion limit violation.

The maximum rod drop time restriction is consistent with the assumed rod drop time used in the accident analyses. Measurement with $T_{avg} \geq 500$ °F and with all reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a reactor trip at operating conditions~~. The surveillance criteria limits for measured drop times are adjusted downward from the TS limit. This ensures that the accident analysis assumptions regarding the negative reactivity insertion rate of the control rod system as a whole remain bounding for a reactor trip concurrent with a postulated seismic event.~~

Insert 2

INSERT 2

The surveillance criteria limits for measured drop times are adjusted downward from the TS limit. This ensures that the accident analysis assumptions regarding the negative reactivity insertion rate of the control rod system as a whole remain bounding. The adjusted limits account for the safety analysis limits and uncertainties. Additionally, measured rod drop time data is trended to ensure that adverse trends in drop time performance are readily identified and addressed.

I. The Additional Conditions contained in Appendix C, as revised through Amendment No. 195, are hereby incorporated into this license. Virginia Electric and Power Company shall operate the facility in accordance with the Additional Conditions.

J. Insert 1

K. ~~J.~~ This license is effective as of the date of issuance and shall expire at midnight on August 21, 2020.

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by:
Harold R. Denton

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Attachment:
Appendices A, B, and C

Date of Issuance: AUG 21 1980

INSERT 1

Consideration of the effects of a concurrent seismic event on the Rod Cluster Control Assembly (RCCA) drop time is excluded from the non-LOCA accident analyses.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original design criteria are met. Misalignment of a rod requires measurement of peaking factors or a restriction in THERMAL POWER; either of these restrictions provides assurance of fuel rod integrity during continued operation. In addition, those accident analyses affected by a misaligned rod are reevaluated to confirm that the results remain valid during future operation.

Continuous monitoring of rod position with respect to insertion limits and rod deviation provided by the rod insertion limit monitor and rod position deviation monitor, respectively. OPERABILITY of the rod position deviation monitor is verified by a functional test at least once per 7 days and by comparison of the indicated positions versus the respective demand position indicators at least once per 12 hours. If the rod position deviation monitor or the rod insertion limit monitor is inoperable, the frequency of manual comparison of indicated rod (or bank) position is increased to an interval of at least once per 4 hours.

In the event that a malfunction of the Rod Control System renders control rods immovable, provision is made for continued operation provided:

- the affected control rods remain trippable, and
- the individual control rod alignment limits are met.

In the event that a malfunction of the Rod Control System renders control rod banks immovable during surveillance testing, provision is made for 72 hours of continued operation provided:

- the affected control rod banks remain trippable,
- the individual control rod alignment limits are met,
- a maximum of one control or shutdown bank is inserted no more than 18 steps below the insertion limit, and
- the SHUTDOWN MARGIN requirements are verified every 12 hours during the period the insertion limit is not met.

Control Bank D is excluded from the 72 hour provision since insertion of D Bank below the insertion limit is not required for control rod assembly surveillance testing.

Checks are performed for each reload core to ensure that bank insertions of up to 18 steps will not result in power distributions which violate the DNB criterion for ANS Condition II transients (moderate frequency transients analyzed in Section 15.2 of the UFSAR) or in a violation of the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 during the repair period.

The 72 hour period for a control rod assembly bank to be inserted below its insertion limit restricts the likelihood of a more severe (i.e., ANS Condition III or IV) accident or transient condition occurring concurrently with the insertion limit violation.

The maximum rod drop time restriction is consistent with the assumed rod drop time used in the accident analyses. Measurement with $T_{avg} \geq 500$ °F and with all reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a reactor trip at operating conditions. ~~The surveillance criteria limits for measured drop times are adjusted downward from the TS limit. This ensures that the accident analysis assumptions regarding the negative reactivity insertion rate of the control rod system as a whole remain bounding for a reactor trip concurrent with a postulated seismic event.~~

Insert 2

INSERT 2

The surveillance criteria limits for measured drop times are adjusted downward from the TS limit. This ensures that the accident analysis assumptions regarding the negative reactivity insertion rate of the control rod system as a whole remain bounding. The adjusted limits account for the safety analysis limits and uncertainties. Additionally, measured rod drop time data is trended to ensure that adverse trends in drop time performance are readily identified and addressed.

Attachment 3

Proposed Unit 1 and Unit 2 Technical Specifications Changes

North Anna Power Station
Units 1 and 2
Virginia Electric and Power Company

UNIT 1

TECH SPEC CHANGE REQUEST NO. 379

TABULATION OF CHANGES

License No. NPF-4 / Docket No. 50-338

Summary of change:

This proposed change to the Technical Specifications is being made to permit elimination of the assumed increase in the RCCA drop time resulting from a concurrent trip and seismic event when determining if the measured rod drop times, including measurement uncertainties, meet the Technical Specification LCO and safety analysis limit of 2.7 seconds.

DELETE	DATED	SUBSTITUTE
License Page 4	03-13-00	License Page 4
B 3/4 1-4a	10-13-99	B 3/4 1-4a

F. The design of the reactor coolant pump and steam generator supports may be revised in accordance with the licensee's submittal dated November 6, 1986 (Serial No. 86-477A).

G. Consideration of the effects of a concurrent seismic event on the Rod Cluster Control Assembly (RCCA) drop time is excluded from the non-LOCA accident analyses.

H. This license is effective as of the date of issuance and shall expire at midnight on April 1, 2018.

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by:
R. C. DeYoung, for

Roger S. Boyd, Director
Division of Project Management
Office of Nuclear Reactor Regulation

Attachments:

1. Construction Related Items to be completed prior to Initial Criticality
2. Appendices A, B, and C
3. Figure 1

Date of Issuance: APR 1 1978

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original design criteria are met. Misalignment of a rod requires measurement of peaking factors or a restriction in THERMAL POWER; either of these restrictions provides assurance of fuel rod integrity during continued operation. In addition, those accident analyses affected by a misaligned rod are reevaluated to confirm that the results remain valid during future operation.

Continuous monitoring of rod position with respect to insertion limits and rod deviation provided by the rod insertion limit monitor and rod position deviation monitor, respectively. OPERABILITY of the rod position deviation monitor is verified by a functional test at least once per 7 days and by comparison of the indicated positions versus the respective demand position indicators at least once per 12 hours. If the rod position deviation monitor or the rod insertion limit monitor is inoperable, the frequency of manual comparison of indicated rod (or bank) position is increased to an interval of at least once per 4 hours.

In the event that a malfunction of the Rod Control System renders control rods immovable, provision is made for continued operation provided:

- the affected control rods remain trippable, and
- the individual control rod alignment limits are met.

In the event that a malfunction of the Rod Control System renders control rod banks immovable during surveillance testing, provision is made for 72 hours of continued operation provided:

- the affected control rod banks remain trippable,
- the individual control rod alignment limits are met,
- a maximum of one control or shutdown bank is inserted no more than 18 steps below the insertion limit, and
- the SHUTDOWN MARGIN requirements are verified every 12 hours during the period the insertion limit is not met.

Control Bank D is excluded from the 72 hour provision since insertion of D Bank below the insertion limit is not required for control rod assembly surveillance testing.

Checks are performed for each reload core to ensure that bank insertions of up to 18 steps will not result in power distributions which violate the DNB criterion for ANS Condition II transients (moderate frequency transients analyzed in Section 15.2 of the UFSAR) or in a violation of the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 during the repair period.

The 72 hour period for a control rod assembly bank to be inserted below its insertion limit restricts the likelihood of a more severe (i.e., ANS Condition III or IV) accident or transient condition occurring concurrently with the insertion limit violation.

The maximum rod drop time restriction is consistent with the assumed rod drop time used in the accident analyses. Measurement with $T_{avg} \geq 500$ °F and with all reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a reactor trip at operating conditions. The surveillance criteria limits for measured drop times are adjusted downward from the TS limit. This ensures that the accident analysis assumptions regarding the negative reactivity insertion rate of the control rod system as a whole remain bounding. The adjusted limits account for the safety analysis limits and uncertainties. Additionally, measured rod drop time data is trended to ensure that adverse trends in drop time performance are readily identified and addressed.

UNIT 2

TECH SPEC CHANGE REQUEST NO. 379

TABULATION OF CHANGES

License No. NPF-7 / Docket No. 50-339

Summary of change:

This proposed change to the Technical Specifications is being made to permit elimination of the assumed increase in the RCCA drop time resulting from a concurrent trip and seismic event when determining if the measured rod drop times, including measurement uncertainties, meet the Technical Specification LCO and safety analysis limit of 2.7 seconds.

DELETE	DATED	SUBSTITUTE
License Page 3	03-13-00	License Page 3
B 3/4 1-4a	10-13-99	B 3/4 1-4a

I. The Additional Conditions contained in Appendix C, as revised through Amendment No. 195, are hereby incorporated into this license. Virginia Electric and Power Company shall operate the facility in accordance with the Additional Conditions.

J. Consideration of the effects of a concurrent seismic event on the Rod Cluster Control Assembly (RCCA) drop time is excluded from the non-LOCA accident analyses.

K. This license is effective as of the date of issuance and shall expire at midnight on August 21, 2020.

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by:
Harold R. Denton

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Attachment:
Appendices A, B, and C

Date of Issuance: AUG 21 1980

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original design criteria are met. Misalignment of a rod requires measurement of peaking factors or a restriction in THERMAL POWER; either of these restrictions provides assurance of fuel rod integrity during continued operation. In addition, those accident analyses affected by a misaligned rod are reevaluated to confirm that the results remain valid during future operation.

Continuous monitoring of rod position with respect to insertion limits and rod deviation provided by the rod insertion limit monitor and rod position deviation monitor, respectively. OPERABILITY of the rod position deviation monitor is verified by a functional test at least once per 7 days and by comparison of the indicated positions versus the respective demand position indicators at least once per 12 hours. If the rod position deviation monitor or the rod insertion limit monitor is inoperable, the frequency of manual comparison of indicated rod (or bank) position is increased to an interval of at least once per 4 hours.

In the event that a malfunction of the Rod Control System renders control rods immovable, provision is made for continued operation provided:

- the affected control rods remain trippable, and
- the individual control rod alignment limits are met.

In the event that a malfunction of the Rod Control System renders control rod banks immovable during surveillance testing, provision is made for 72 hours of continued operation provided:

- the affected control rod banks remain trippable,
- the individual control rod alignment limits are met,
- a maximum of one control or shutdown bank is inserted no more than 18 steps below the insertion limit, and
- the SHUTDOWN MARGIN requirements are verified every 12 hours during the period the insertion limit is not met.

Control Bank D is excluded from the 72 hour provision since insertion of D Bank below the insertion limit is not required for control rod assembly surveillance testing.

Checks are performed for each reload core to ensure that bank insertions of up to 18 steps will not result in power distributions which violate the DNB criterion for ANS Condition II transients (moderate frequency transients analyzed in Section 15.2 of the UFSAR) or in a violation of the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 during the repair period.

The 72 hour period for a control rod assembly bank to be inserted below its insertion limit restricts the likelihood of a more severe (i.e., ANS Condition III or IV) accident or transient condition occurring concurrently with the insertion limit violation.

The maximum rod drop time restriction is consistent with the assumed rod drop time used in the accident analyses. Measurement with $T_{avg} \geq 500$ °F and with all reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a reactor trip at operating conditions. The surveillance criteria limits for measured drop times are adjusted downward from the TS limit. This ensures that the accident analysis assumptions regarding the negative reactivity insertion rate of the control rod system as a whole remain bounding. The adjusted limits account for the safety analysis limits and uncertainties. Additionally, measured rod drop time data is trended to ensure that adverse trends in drop time performance are readily identified and addressed.

Attachment 4

Significant Hazards Consideration Determination

North Anna Power Station
Units 1 and 2
Virginia Electric and Power Company

SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Virginia Power is proposing a risk informed license condition that permits elimination of the assumed increase in the rod control cluster assembly (RCCA) drop time resulting from a concurrent trip and seismic event when determining if the measured rod drop times including measurements uncertainties, meet the Technical Specification LCO and safety analysis limit of 2.7 seconds.

The proposed change does not alter the limiting results of the safety analyses presented in Chapter 15 of the Updated Final Safety Analysis Report (UFSAR). Safety margins are maintained. The proposed change will not alter the ability of the reactor protection and control system to perform their design functions or to meet the applicable criteria set forth in the Institute of Electrical and Electronic Engineers (IEEE) and American Nuclear Standard Institute (ANSI) standards and in 10 CFR 50 Appendix A. Analyses continue to demonstrate that the RCCAs will perform their design function by inserting into the core following a reactor trip, even with a postulated concurrent seismic event. However, a postulated seismic event concurrent with an event leading to reactor trip could lead to control rod insertion times slightly in excess of those assumed in the accident analysis. The proposed change results in an insignificant increase in core damage frequency or risk. However, the proposed change is conservatively judged to be a malfunction of a type different than already analyzed and therefore, an unreviewed safety question.

Virginia Electric and Power Company has reviewed the requirements of 10 CFR 50.92 as they relate to the proposed changes for the North Anna Units 1 and 2 and determined that a significant hazards consideration is not involved. The following is provided to support this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated

   Elimination of the assumed increase in the RCCA drop time resulting from a concurrent trip and seismic event when determining if the measured rod drop times, including measurement uncertainties, meet the accident analysis limit does not contribute to the probability of previously analyzed accidents. The proposed change will not alter the limiting results of the safety analyses presented in Chapter 15 of the UFSAR. Although the proposed change eliminates an accident consideration that is currently addressed in the UFSAR accident analyses (i.e. any Chapter 15 accident with the effects of a concurrent seismic occurrence reflected in the RCCA drop time), there is no significant increase in the probability or consequences of any accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated

   There are no modifications to the plant as a result of the changes. No new accident or event initiators are created by eliminating the assumed increase in the RCCA drop time resulting from a concurrent trip and seismic event. The

proposed change will not alter the ability of the reactor protection and control system to perform their design functions or to meet the applicable criteria set forth in the IEEE and ANSI standards and in 10 CFR 50 Appendix A. Therefore, the proposed changes do not create the possibility of any accident or malfunction of a different type previously evaluated.

3. Does the change involve a significant reduction in a margin of safety

   The proposed change will not alter the limiting results of the safety analyses presented in Chapter 15 of the UFSAR. Elimination of the assumed increase in the RCCA drop time resulting from a concurrent trip and seismic event when determining if the measured rod drop times, including measurement uncertainties, meets the accident analysis limit maintains adequate safety margin in the safety analysis. Therefore, the proposed change does not significantly reduce a margin of safety.