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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 32 TO

FACILITY OPERATING LICENSE NO. NPF-4

VIRGINIA ELECTRIC AND POWER COMPANY

NORTH ANNA POWER STATION, UNIT NO. 1

DOCKET NO. 50-338

Introduction:

By letters dated October 15 and December 23, 1980, the Virginia Electric and Power Company (the licensee) proposed changes to the Technical Specifications for Facility Operating License NPF-4 for the North Anna Power Station, Unit No. 1 (NA-1).

The changes involve: (1) requirements for providing redundancy in decay heat removal capability; (2) addition of certain TMI-2 Lessons Learned Category "A" requirements to the NA-1 Technical Specifications; and (3) requirements for the maintenance of a minimum water level above fuel assemblies.

The licensee's proposed changes as noted above are in response to NRC letters dated (1) June 11, 1980; (2) July 2, 1980; and (3) August 15, 1980.

The above items were addressed as part of our review of the North Anna Power Station, Unit 2 (NA-2) prior to issuance of a full power operating license for this facility. And, these items were incorporated in the Appendix A Technical Specifications to the NA-2 full power Facility Operating License NPF-7 issued on August 21, 1980.

Where applicable, the above requirements are being incorporated in the NA-1 Technical Specifications in a manner identical to the already existing, approved NA-2 Technical Specifications. This fact is so indicated, where appropriate, in our evaluation which follows.

Discussion and Evaluation:

(1) Decay Heat Removal Capability

Our letter of June 11, 1980 to all operating pressurized water reactor licensees requested that the licensees submit revised Technical Specifications (TS) regarding decay heat removal capability. Our request was based on a number of events which had occurred at operating pressurized

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water reactors where decay heat removal capability was degraded due to inadequate administrative controls in effect at the time the facilities were in shutdown modes of operation.

The licensee has proposed changes which would require Residual Heat Removal (RHR) Subsystems A and B be specified as operable in Modes 4 and 5 (Hot Shutdown and Cold Shutdown) with operability determined by the surveillance requirements specified in TS 4.0.5.

For Mode 6 (Refueling), the proposed changes specify that one RHR loop be in operation during Mode 6 to ensure that (1) sufficient cooling capacity is available to remove decay heat in Mode 6, and (2) adequate flow is maintained through the reactor core to minimize the effect of a boron dilution event and to prevent boron stratification. In addition, the RHR loop is determined operable by circulating reactor coolant at a flow rate of greater than or equal to 3000 gallons per minute at least once per 4 hours.

Finally, Limiting Conditions for Operation (LCO) in Mode 6, as specified in the proposed changes for TS 3.9.8.2 requires that two independent RHR loops be in an operable status.

Based on the above, we find that the proposed changes meet our concerns as stated in our June 11, 1980 letter for RHR redundancy. Also, the changes are indentical to the existing, approved NA-2 TS. Therefore, we find these changes to be acceptable.

(2) TMI-2 Lessons Learned Category "A" Items

The requirements established from our review of the TMI-2 accident were issued to all operating nuclear power plants on September 13, 1979. Some of these requirements were designated as Lessons Learned Category "A" requirements and were to be completed by the licensee not later than January 31, 1980. Our evaluation of the licensee's compliance with the Category "A" items was enclosed in our letter to the licensee dated April 23, 1980.

By letter dated July 2, 1980, we requested that the licensee amend the NA-1 TS to incorporate LCO and Surveillance Requirements for the Category "A" items. Included in our request were model TS which we had determined to be acceptable.

The licensee's letters dated October 15 and December 23, 1980 are in direct response to our July 2, 1980 request. Each of the Category "A" issues as identified by the NRC staff and the licensee's response are provided below.

2.1.1 Emergency Power Supply Requirements

Pressurizer water level indicators, pressurizer relief and block valves, and pressurizer heaters provide an important function in a post-accident condition. Adequate emergency power supplies for these components provide additional assurance of post-accident operability. As stated in our Safety Evaluation dated April 23, 1980, the licensee has provided the required emergency power supplies for these components.

For the pressurizer water level indication, the presently existing NA-1 TS provide acceptable surveillance and LCO in the event of component inoperability.

The licensee has proposed changes to the NA-1 TS for the pressurizer relief and block valves and the pressurizer heaters. The licensee's changes will provide a 31 day channel check and an 18 month channel calibration check and provide LCO in the event of component inoperability.

We have reviewed the licensee's proposed changes and find that the emergency power supplies are reasonably ensured for post-accident functioning of the subject components. Also, the proposed changes are identical to the existing, approved NA-2 TS. Therefore, we find the proposed changes to be acceptable.

2.1.3.a Direct Indication of Flow (Valve Position)

The licensee has provided direct indication of power-operated relief valves (PORVs) and safety valve position indication in the control room. Also, pressure and temperature sensors downstream of the PORVs and safety valves provide backup methods for determining the position of the valves. These indications are a diagnostic aid for the plant operator and provide no automatic action.

The licensee has proposed changes to the NA-1 TS which would provide a 31 day channel check and an 18 month channel calibration check and provide LCO in the event of component inoperability. The licensee's proposed changes meet our July 2, 1980 criteria and are identical to the existing, approved NA-2 TS. Therefore, we find the proposed changes to be acceptable.

2.1.3.b Instrumentation for Inadequate Core Cooling

The licensee has installed an instrument system to detect the effects of low reactor coolant level and inadequate core cooling. These instruments, subcooling meter, receive and process data from existing plant instrumentation. We have previously reviewed this system in our Safety Evaluation dated April 23, 1980 and found the system to be acceptable.

The licensee has proposed changes which will provide for a 31 day channel check and an 18 month channel calibration with appropriate LCO in the event of instrument inoperability. We find the licensee's proposed changes are in conformance with our July 2, 1980 criteria and are identical to the existing, approved NA-2 TS. Therefore, we find the proposed changes to be acceptable.

2.1.4 Diverse Containment Isolation

The licensee has categorized all NA-1 systems penetrating containment as being either essential or non-essential. All non-essential systems having automatic containment isolation valves and not required for an orderly reactor shutdown or to maintain containment atmospheric conditions are closed on Phase A isolation which is initiated by a safety injection actuation signal. Safety injection is activated by the following diverse parameters: (1) high containment pressure, (2) low pressurizer pressure; (3) high steam line differential pressure; (4) high steam line flow with either low steam line pressure or low-low T average, and (5) manual activation.

Essential systems are divided into two categories (levels) which are based on their ability to mitigate the severity of various types of accidents. Level 1 essential systems are defined as Engineered Safety Features and Containment Depressurization Systems and are required to operate after a loss-of-coolant accident (LOCA).

Essential Level 2 systems are defined as those required to maintain the operation of critical systems and functions such as containment heat removal and, therefore, remain unisolated from the containment until a design basis LOCA is indicated (Phase B isolation).

The licensee identified several non-essential systems that are not automatically isolated by containment isolation signals. However, as stated in our April 23, 1980 Safety Evaluation, we have determined adequate isolation provisions have been provided for all non-essential systems.

The NA-1 containment isolation system will not result in the automatic loss of containment isolation after the conta: ent isolation signal is reset. Reopening of containment requires deliberate operator action and there are no valve control switches which control the reopening of more than one isolation valve.

The NA-1 TS Table 3.6-1 presently lists each affected containment isolation valve with appr iate surveillance testing and LCO in the event of valve inoperability and therefore meets the criteria specified in our July 2, 1980 letter. No further actions are required and item 2.1.4 is complete.

2.1.7.a Auto Initiation of Auxilary Feedwater System

The auxilary feedwater system for NA-1 & 2 was designed as a safety-related system, aside and apart from any TMI-2 related requirements. Consistent with the design intent, the auxiliary feedwater system initiating circuitry for NA-1 & 2 incorporates both automatic and manual system start capability, including manual initiation of the system from the control room. Manual initiation capability is provided independent of automatic initiation, and the design of the automatic initiation circuitry is such that a single-failure will not result in the total loss of the auxiliary feedwater system.

Based on the criteria specified in our July 2, 1980 letter, the licensee has proposed changes to the NA-1 TS to incorporate LCO and surveillance testing for (1) manual initiation circuitry and

(2) automatic actuation logic required to test auto-initiating circuitry. A channel functional test of manual initiation will be required on a 31 day basis and a bi-monthly test will be required for the automatic actuation logic.

The licensee has proposed changes which will require that the auxiliary feedwater system be demonstrated operable prior to entry into Mode 3 (Hot Standby) following each cold shutdown (Mode 5). Operability will be determined by performing a flow test to verify the normal flow path from the emergency condensate storage tank through each auxiliary feedwater pump to its associated steam generator.

The licensee's proposed changes list the appropriate auxiliary feedwater components, provide proper surveillance frequencies, and LCO in the event of component inoperability. Also, the proposed changes are in conformance with our July 2, 1980 criteria and are identical to the existing, approved NA-2 TS. Therefore, we find the proposed changes to be acceptable.

2.1.7.b Auxiliary Feedwater Flow Indication

The licensee has installed auxiliary (emergency) flow indication to the steam generators which is indicated in the control room. The flow indication meets our vital power and testability requirements as stated in our Safety Evaluation dated April 23, 1980.

The licensee has proposed changes to the NA-1 TS which will provide for a 31-day channel check and an 18-month channel calibration with the appropriate LCO in the event of instrument inoperability. We find the licensee's proposed changes meet our July 2, 1980 criteria and are identical to the existing, approved NA-2 TS. Therefore, we find the proposed changes to be acceptable.

2.2.1.b Shift Technical Advisor (STA)

Our July 2, 1980 request indicated that the NA-1 TS related to minimum shift manning should be revised to reflect the addition of one STA to each shift to perform the function of accident assessment.

This required revision to the NA-1 TS was completed with the issuance of Amendment No. 19 to the NA-1 Facility Operating License NPF-4 on August 5, 1980. No further actions are required and item 2.2.1.b is complete.

License Conditions for TMI-2 Lessons Learned Category "A" Items

Our letter dated July 2, 1980 indicated the NA-1 Facility Operating License NPF-4 should be amended by adding certain license conditions as recommended from our review of the TMI-2 accident and as specified in NUREG-0578, TMI-2 Lessons Learned Task Force Status Report and Short Term Recommendations, July 1979. These items are addressed below.

2.1.6.a Systems Integrity (Outside Containment)

Our letter dated July 2, 1980 indicated that the NA-1 license should be amended by adding a license condition related to a Systems Integrity Measurements Program. Such a condition requires the licensee to effect an appropriate program to eliminate or prevent the release of significant amounts of radioactivity to the environment by way of leakage from the engineered safety systems and auxiliary systems, which are located outside containment.

Our review and evaluation of the licensee's System Integrity Measurements Program is provided in our Safety Evaluation dated April 23, 1980. Facility systems specified in the program include recirculation spray, safety injection, chemical and volume control, gas stripper and the hydrogen recombiners.

By letter dated October 15, 1980 the licensee proposed changes to the NA-1 TS which would include a Systems Integrity Program as part of TS 6.8, Procedures and Programs. The licensee's program as stated in TS 6.8.4.a is identical to the criteria specified in our July 2, 1980 letter and the proposed changes are identical to the existing, approved NA-2 TS. Therefore, we find the proposed changes to be acceptable.

2.1.8.c Iodine Monitoring

Our letter dated July 2, 1980 indicated that the NA-1 license should be amended by adding a license condition related to iodine monitoring. Such a condition would require the licensee to implement a program which would ensure a capability to determine airborne iodine concentrations in those areas requiring personnel access under accident conditions.

Our evaluation of the licensee's improved in-plant Iodine Monitoring Program is provided in our Safety Evaluation dated April 23, 1980, wherein we state that the program has the capability to accurately monitor iodine in the presence of other noble gases.

By letter dated October 15, 1980 the licensee proposed changes to the NA-1 TS which would include an Iodine Monitoring Program as part of TS 6.8, Procedures and Programs. The proposed changes as stated in TS 6.8.4.b are identical to the criteria specified in our July 2, 1980 letter, and the proposed changes are identical to the existing, approved NA-2 TS. Therefore, we find the changes to be acceptable.

2.1.3.b Backup Method for Determining Subcooling Margin

In our letter of July 2, 1980 we indicated that the NA-1 license be amended (Optional) by adding a license condition related to the accurate determination of reactor coolant system subcooling margin. The prompt monitoring of the subcooling margin can provide warning of inadequate core cooling. Our letter indicated the license condition should provide procedures to accurately monitor the subcooling margin.

By letter dated October 15, 1980 the licensee proposed changes to the NA-1 TS 3/4.4, Reactor Coolant System, by incorporating revised LCO and bases to ensure both adequate coolant flow and that reactor coolant temperature and pressure are maintained or are adjusted to achieve an appropriate subcooling margin. The proposed changes are identical to the existing, approved NA-2 TS (except for overpressure protection provisions) and therefore, we find the changes to be acceptable.

II.K.3 Reporting Requirements For All Challenges to PORVs and Safety Valves

By letter dated December 23, 1980 the licensee proposed changes to the NA-1 TS which would require the documentation of PORVs and Safety Valves on a monthly basis. The requested change is in compliance with NUREG-0660, The TMI Action Plan, Section II.K.3. The documentation will provide an operational data base on challenges to relief and safety valves at NA-1.

The requested change would revise the NA-1 TS, 6.9.1.6, Monthly Operating Report, to include the specifications of Section II.K.3, NUREG-0660. Also, the revised change is identical to the existing, approved NA-2 TS. Therefore, we find this change to be acceptable.

3. Minimum Water Level for Refueling Operations

Our letter of August 15, 1980 to all Westinghouse pressurized water reactor licensees requested that the licensee review the NA-1 TS and Procedures to assure that fuel assemblies and control rods not be inadvertently exposed during refueling operations. Our letter specifically requested that a minimum of 23 feet of water be maintained over the reactor vessel flange in place of the then specified Westinghouse Standard Technical Specification for maintaining 23 feet of water over fuel assembles in the core.

By letter dated September 16, 1980 the licensee stated that the present Operating Procedure, OP-4.1, Refueling and Fuel Handling for NA-1 & 2, requires a level of 27 feet plus or minus 6 inches of water be maintained above the reactor vessel flange. We find this procedure to be acceptable and conservative based on the request as specified in our August 15, 1980 letter.

By letter dated October 15, 1980, the licensee proposed changes to the NA-1 TS for the maintenance of a minimum water level of at least 23 feet of water over the top of the reactor pressure vessel flange. The licensee's proposed change is in conformance with the requirements specified in our August 15, 1980 letter and is identical to the existing, approved requirement specified in the NA-2 TS. Therefore, we find the proposed change to be acceptable.

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 §51.5(d)(4), that an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of the amendment.

CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) because this amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment 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.

Date: June ?, 1981