ENCLOSURE 2

# NORTH ANNA UNITS 1 & 2
# 582.8°F REACTOR COOLANT SYSTEM
# STONE & WEBSTER/BOP
# SAFETY EVALUATION SUMMARY

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APRIL 1982

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A. OBJECTIVE

To provide a technical basis for determining that the proposed 2.5°F increase in reactor coolant system Tavg does not involve an unreviewed safety question in accordance with the requirements of 10CFR50.59. This review is limited to systems within Stone & Webster Engineering Corporation's original scope of work. The NSSS and Turbine-Generator review has been performed by Westinghouse and is documented as Enclosure 1.

Our evaluation used the following parameters which bound or are equivalent to the proposed uprated conditions:

| | |
|---|---|
| Main Steam Pressure 100% Power | 900 psia |
| Main Steam Temp. No-Load | 547°F |
| Main Steam Pressure No-Load | 1020 psia |
| RCS Tavg | 582.8°F |
| Steam Flow $10^6$ lb./hr Total | 12.12 |
| Reactor Power $MW_t$ | 2775. |
| NSSS Power $MW_t$ | 2785. |

B. CONCLUSION

The proposed change in reactor coolant system average temperature has been reviewed and evaluated with respect to the following:

1. Accident Analysis and Environmental Qualification
2. Pipe Stress and Supports
3. Major Equipment Pipe Rupture Restraints and Equipment Supports
4. BOP System and NSSS Interfaces
5. Review of Technical Specifications

Based on the results of our review it has been concluded that the proposed 2.5°F increase in Tavg does not represent an unreviewed safety question as defined in 10CFR50.59. The summary of the analyses related to the above are attached.

1. It has been determined that the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety and previously evaluated in the Safety Analysis Report has not been increased.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report has not been created.

3. The margin of safety as defined in the basis for any Technical Specification has not been reduced.

C. ACCIDENT ANALYSIS

1. Containment Loss of Coolant Accident

The present licensed power level for North Anna 1 & 2 is 2775 MWT (2785 including reactor coolant pump heat). The analyses for containment integrity, containment depressurization, low-head safety injection pump NPSH and recirculation spray pump NPSH for the 2.5°F uprate are bounded by the present analyses which are based on core power of 2900 MWT (2910 MWT including reactor coolant pump heat).

2. Containment Main Steam Line Break Analysis

The basis of the steam line break analysis is the full guillotine main steam line break at no-load (hot shutdown) condition. The no-load Tavg of 547°F remains unchanged subsequent to the uprate and therefore the Main Steam Line Break conditions remain as previously analyzed.

3. Subcompartment Analysis

Subcompartment analyses were performed and documented in the FSAR for the reactor cavity, steam generator cubicle and pressurizer cubicle. For the subcooled reactor coolant system, mass and energy releases decrease with increased reactor coolant temperature. The analyses documented in the FSAR are therefore bounding for the uprate.

4. Equipment Qualification

1. Inside Containment:

Equipment qualification inside the containment is based on the Main Steam Line Break and LOCA post-accident environmental conditions. The current Steam Line Break and LOCA analyses are bounding for the uprate conditions as discussed in sections 1 & 2.

2. Outside Containment:

Post-accident environments outside containment which are used to generate equipment qualification envelopes are based on the following high energy line breaks:

a. Primary System Branch Line Break

The Letdown Line Break is part of the basis for the environmental qualification in the charging pump cubicle in the auxiliary building. The letdown line temperature increases one degree above the normal operating temperature used in the original analysis. The resulting environmental temperature due to a postulated line break is not significantly affected by this change.

2. Outside Containment-(cont'd.)

b. Secondary System Break

The Main Steam Line Break affects the Main Steam Valve House, the Service Water Valve Pit and the Turbine Building. The Main Steam Valve House environmental envelope is based on no-load power condition which is unaffected by the uprate. Equipment qualification temperature and pressure in the Service Water Valve Pit and Turbine Building is limited by the Turbine Building siding pressure retaining capability. Any change to break effluent due to the uprate has no effect on this pressure or temperature and, therefore, on equipment qualification.

c. Auxiliary Steam Line Break:

This affects the Auxiliary Building and the Service Water Valve Pit. The releases are based on the Auxiliary Steam Line relief valve pressure setting which is unchanged by the uprate. Additionally, the Auxiliary Steam System pressure is controlled by a pressure reduction valve tied into the Main Steam header. The increased Main Steam operating pressure will not affect the operation of this valve and therefore the Auxiliary Steam System pressure will remain unchanged.

d. Steam Generator Blowdown Line Break:

This affects the Pipe Tunnel and the Auxiliary Building. The releases are based on the bounding condition of no-load steam generator pressure which is unchanged by the uprate.

D. PIPE STRESS AND SUPPORTS

All piping systems directly affected by the 2.5°F Reactor Coolant System Tavg uprate were reviewed with respect to pipe stress and the adequacy of pipe support designs. The systems most obviously impacted were the Main Steam and Feedwater Systems. Pipe stress and support calculations for these systems were reviewed and it was determined that the uprate would have an impact on three Unit 2 Main Steam Mono-Ball supports located on a non-safety related portion of piping. These supports will be modified to accommodate the uprated conditions. All the remaining piping and supports on the Main Steam System and other systems were reviewed at the bounding no-load condition or were evaluated to confirm adequate margins existed within the calculations to accommodate the uprated conditions.

The piping connected to the reactor coolant loops was reviewed to determine the effect of a possible increase in displacements resulting from increased Reactor Coolant system temperature. The increase in Reactor Coolant System temperature is less than one percent and therefore the additional displacement is considered to be insignificant. Therefore no reanalysis is necessary and these systems are acceptable.

D. PIPE STRESS AND SUPPORTS -(cont'd.)

A review of safety related piping for fatigue effects of thermal transients revealed that only the reactor coolant letdown line required reanalysis. The 2.5°F increase in Tavg subjected the letdown line to a more severe reinitiation of letdown flow transient than was originally used as a design basis. The results of the analysis showed that the fatigue effects on the piping would remain acceptable subsequent to the proposed uprate. The stress report will be revised to include this information.

It has been determined that previously calculated break points in piping systems, used for the design of rupture restraints, jet shields and large pipe supports will remain unchanged as a result of the uprating. The points of maximum stress within any piping system remain unchanged as the stress increases due to the uprating are uniform. Pipe rupture effects are discussed in Section E.

E. MAJOR EQUIPMENT PIPE RUPTURE RESTRAINTS AND EQUIPMENT SUPPORTS

Design calculations for major equipment supports, seismic tanks, vessels, pipe rupture restraints and shields and miscellaneous mechanical equipment have been reviewed with the uprated system parameters for 100% power and no-load conditions.

Of the approximately 450 calculations reviewed, five were found to have used Main Steam or Feedwater System pressures that would not bound values expected subsequent to the 2.5°F Reactor Coolant System Tavg increase. These five calculations were reviewed with respect to the uprated conditions and found acceptable.

Our evaluation concluded that an increase in Reactor Coolant System temperature results in a decrease in LOCA loadings with the frequency content of those loadings being unchanged as a result of the uprating. This conclusion was based on the information contained in the Westinghouse safety evaluation, Enclosure 1.

## F. BOP SYSTEMS AND NSSS INTERFACES

### 1. Condensate and Feedwater Systems

Steam conditions leaving the turbine are unchanged. As a result of increasing Tavg, the Steam Generator Pressure will not exceed the design pressure and temperature of the Condensate and Feedwater Systems. Equipment such as steam generator feed pumps, condensate pumps, heater drain pumps and feedwater heaters will continue to operate unchanged since design steam flows are greater than those expected subsequent to the uprating.

A review of the operation of the Main Feedwater Regulating and Bypass Valves has shown that increasing Tavg by 2.5°F will not cause a significant change in current operating flexibility.

### 2. Main Steam System

The Main Steam System piping and components are designed for no-load conditions and are bounded by the proposed 2.5°F Reactor Coolant System Tavg increase.

The operability of the Main Steam Trip Valves (MSTV's) and Non-Return Valves (NRV's) has been reviewed with regard to the uprated conditions. The review has shown that the MSTV's and NRV's will perform adequately at the uprated conditions. The previous analyses used parameters that bounded the uprated conditions. The structural adequacy of the NRV's and MSTV's was evaluated for impact loadings and was found to be acceptable for the uprated conditions as the analyses used conditions which bounded the uprate.

Because the Main Steam System transients will remain unchanged as a result of the uprating, the existing Main Steam Safety Valves are adequately sized for the uprated conditions and will not require revised lift settings.

### 3. Component Cooling Water System

The increased Reactor Coolant System temperature increases the heat loadings to various Chemical and Volume Control System (CVCS) heat exchangers and therefore the Component Cooling Water System will be required to remove a small amount of additional heat subsequent to the uprate. A review has shown that adequate margin exists in the Component Cooling Water System to remove the additional heat load.

4. Steam Generator Blowdown System

A review of the entire Steam Generator Blowdown System has indicated that uprating Reactor Coolant System Tavg by 2.5°F will not affect the present safety aspects or operability of the system.

The design of the excess flow high energy line break isolation valves was for an inlet pressure of 1100 psig which is higher than the lowest Main Steam Safety Valve setpoint and is therefore acceptable with regard to the uprate.

All remaining portions of the Steam Generator Blowdown System including flow control valves, safety valves, tanks and pressure control valves were reviewed for any expected temperature and pressure changes and are unaffected by the uprate.

5. Auxiliary Feedwater System

The Auxiliary Feedwater System is designed to remove decay and sensible heat from the Reactor Coolant System following a reactor trip. The original Auxiliary Feedwater System requirements were based on an NSSS rating of 2910 $MW_t$ and 850 psia steam pressure. The 2.5°F Reactor Coolant System Tavg increase at 2785 $MW_t$ is bounded by the original analysis that included the 2.5°F Tavg increase in Reactor Coolant System temperature. Therefore, the Auxiliary Feedwater System requirements are unchanged and the Auxiliary Feedwater Pumps are of adequate capacity and head to supply the required flow at the uprated conditions.

G. REVIEW OF THE TECHNICAL SPECIFICATIONS

The Technical Specifications have been reviewed to determine if any sections could be affected by the proposed 2.5°F Tavg increase from 580.3°F to 582.8°F. With the exception of the Technical Specification revisions recommended by Westinghouse Electric Corporation in their safety evaluation, Enclosure 1, no additional sections are affected.