THE B&W NSS CHANGE

ASSESSMENT PROGRAM (CAP)

FOR PLANT IMPROVEMENTS

FOR

VIRGINIA ELECTRIC & POWER COMPANY

NORTH ANNA UNIT NO. 3

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BY

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I. INTRODUCTION

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With the reactivation of North Anna 3 in January 1981, Vepco requested B&W to review the design basis for the NSS and other systems and equipment supplied by B&W and to provide recommendations for changes commensurate with a planned commercial operation of 1989.

B&W conceived a "Change Assessment Program" (CAP) to systemmatically review each fluid system, major pieces of equipment, and I&C designs as they existed when equipment was built. This document briefly describes the background for the CAP, those items which are actively being pursued for potential implementation into the plant design, and those items for which reviews are being planned.

2. THE CHANGE ASSESSMENT PROGRAM

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The Change Assessment Program (CAP) is an outgrowth and an expansion of a similar program used on the B&W 205 FA plants. The CAP program is a formal method for keeping track of potential changes to equipment and systems as a result of new, revised or newly imposed Regulatory Guides, NUREGS, operating plant experience and concerns. The CAP program is also as forward looking as practical by considering potential future licensing and plant operational requirements.

This CAP program technique is being applied to North Anna Unit 3 by using the PSAR (and the Regulatory Guide Positions addressed therein) as a baseline design. The CAP list of potential modifications was developed by reviewing the categories of documents noted above added to the TMI-2 upgrades identified in NUREG 0737.

The following section lists the Potential Modifications which are under consideration by VEPCo.

3. POTENTIAL MODIFICATIONS

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3.1 Secondary Plant Improvements

3.1.1 Modulating Atmosphere Dump Valve Control

Redundant safety grade modulating atmospheric dump valve and controls are to be provided on each steam generator.

This change will provide single failure proof steam generator pressure control to allow for controlling decay heat removal via the steam generator to achieve cold shutdown conditions.

3.1.2 AFW Backup Source

A safety grade backup source of AFW will be supplied from sources such as Lake Anna or the Service Water Reservoir. The use of the backup source will be an aid in meeting the requirements of RG 1.139

3.1.3 Secondary System Heatup

With the existing heatup system, the high elevations of the main feedwater lines and heatup circulation pumps while in the operating temperature range cause flashing in the main feedwater lines. This condition will be prevented by a steam generator vacuum heatup procedure coupled with a forward flushing of the main feedwater lines, during the plant heatup process, to resolve the flashing problem and still comply with the steam generator temperature limits.

3.1.4 Startup Feedwater Valve Bypass For Low Flow

During cooldown, heatup and hot standby operations main feedwater must be controlled to a low flow value. The feedwater system is being reviewed to provide flow control in the low range during startup and shutdown.

3.1.5 Modulating Condenser Dump Bypass For Low Flow

During hot standby operations, with no core decay heat, the only heat input to the secondary system is from the reactor coolant pumps. In order to remain at a constant steam pressure, the modulating condenser dump (MCD) valves must be able to control down to a steam flow equivalent to the heat input from one to four reactor coolant pumps. A modification to the Control System is being considered so that only one of the four (MCD) valves will be able to modulate for control of low flow.

3.1.6 Spurious Opening of Turbine Bypass Valve

North Anna Unit 3 is provided with 105% turbine bypass valve capacity. A large portion if not all of this capacity could be inadvertently actuated due to failure in the control system, resulting in an overcooling event for the reactor core. One method being considered to reduce the severity of this event is the interlocking of the turbine bypass valve banks so that no more than one bank can inadvertently open or fail to open, when required, on a spurious failure.

3.1.7 Reactor Trip on Low OTSG Pressure

A safety grade low steam pressure reactor trip is planned to be provided which would lessen the RC pressure transients for overcooling events that deplete steam generator invertory.

3.1.8 Main Feedwater Overfill Prevention

A safety grade main feedwater steam generator overfill prevention instrumentation will be provided. The Control System can be modified to limit main feedwater overfill of a steam generator by isolating MFW to the affected steam generator. The modification is based on a main feedwater flow/neutron flux ratio for mid to high reactor power operation and a wide range steam generator Delta-P/Neutron flux ratio for startup through low reactor power operation and upon reactor trip.

3.1.9 Reduced Main Feedwater Flow on Reactor Trip

The need for possible modifications to achieve smooth reduction of feedwater following a reactor trip will be evaluated.

3.1.10 Anticipatory Reactor Trip on Loss of Feedwater

A safety grade anticipatory reactor trip on loss of feedwater is being evaluated. The loss of the feedwater trip under consideration would utilize a neutron flux to main feedwater flow function to determine when main feedwater flow is less than the minimum required for a given power level.

3.1.11 Feed Only Good Generator (FOGG)

The FOGG logic is designed to detect which steam generator is capable of controlled heat removal following steam line or feedwater line breaks and to isolate main and auxiliary feedwater to the steam generator with the break.

3.1.12 Auxiliary Feedwater Control System Upgrades

Auxiliary Feedwater (AFW) flow control will be provided with safety grade instrumentation and control valves.

Auxiliary feedwater control will be provided with three steam generator level control setpoints. The setpoints correspond to the following detected events classified as modes: A) Loss of main feedwater with reactor coolant pumps running, B) Natural circulation with main feedwater and reactor coolant pumps not running, C) Loss of coolant accident. High and low level ramp rates would be provided for Mode C. The mode and ramp rates are selected to provide sufficient cooling for the RCS and to preclude overcooling.

The loss of main feedwater event is detected by low SG level, loss of main feedwater pumps, loss of four RC pumps, or flux/main feedwater reactor trip. The LOCA mode is selected on low RC pressure or high reactor building pressure coincident with loss of RCS subcooling.

3.1.13 MFW Reliability Improvements

A review will be performed to evaluate the reliability of the MFW System.

3.2 Reactor Coolant System

3.2.1 Saturation Margin Equipment

Safety grade RCS saturation margin equipment will be provided by:

1. Providing displays for inadequate core cooling in the control room.
2. Providing signals to trip the reactor coolant pumps and to signal a LOCA condition to the AFW controls for mode selection.

The saturation margin equipment utilizes the RC pressure signal and the extended range RCS T_{hot} signal for inputs to determine RCS saturation margin.

3.2.2 Direct Indication of Relief and Safety Valve Position

Pressure relief of the Reactor Coolant System is provided by the pilot operated relief valve (PORV) and safety valves. PORV position indication should be provided in the control room.

3.2.3 PORV, Controls and Block Valve Upgrades

Upgrades for the PORV, controls and block valve will include:

3.2.3.1 Replace the PORV with a qualified valve having direct position indication.

3.2.3.2 Upgrade PORV controls.

3.2.3.3 Replace the PORV block valve with a qualified valve.

3.2.3.4 Provide safety grade signals to PORV block valve to close on coincident low RC pressure and PORV open position.

3.2.3.5 Perform analysis to determine automatic PORV block valve closure setpoint.

3.2.3.6 Evaluate use of additional PORV's and associated block valves.

3.2.4 Larger Pressurizer

The pressurizer will be enlarged by approximately 40 percent to improve operability.

3.2.5 Flux Trip From Overcooling Transients

Process induced nuclear instrumentation errors during small overcooling and steam line breaks delay or prevent the flux signal related reactor trips (overpower, flux/flow, DNBR, offset). Process induced errors are a result of reactor vessel downcomer water temperature decreases during overcooling/SLB transients. The decreasing water temperature lessens the neutron flux incident on the power range detectors.

This condition can be corrected through an alternate power measurement from the flux measurement. This involves the use of existing measurements of RC inlet and outlet temperatures. This hardware correction will be used if analysis does not eliminate the concern.

3.2.6 RC System Sampling

A safety grade sampling system will be provided on the RCS to determine the radiological and chemical composition of the reactor coolant. The sampling equipment will be available and accessible during post accident conditions. This equipment will also provide a method for monitoring boric acid concentration during periods of natural circulation and other operating modes.

3.2.7 High Point Vents

High point vents on the hot legs and RV head will be provided and will be remotely operated, seismically qualified, environmentally qualified, and provided with positive position indication in the control room.

3.2.8 Pressurizer Controls Upgrade

Two safety grade pressurizer level monitors will be provided. The pressurizer level monitors will be compensated for the density of the pressurizer fluid and the density of the reference leg fluid. The compensated signal will be utilized during normal, abnormal and accident conditions.

3.2.9 Reactor Vessel Level

A level measurement system will be evaluated for the reactor coolant system to be used in monitoring for inadequate core cooling.

3.3 Makeup and Purification System

3.3.1 High Pressure Auxiliary Pressurizer Spray

Provide a high pressure spray to the pressurizer from the makeup pumps. During a RC system cooldown event the normal pressurizer spray will not be available if the RC pumps are not operable.

An auxiliary spray line from pressurizer spray will be provided that originates from the makeup pump discahrge line.

3.3.2 HPI System Which Can Be Throttled

The HPI System is required to operate after an ECCS accident in order to control HPI flow into the RCS. A throttled HPI flow is being considered for improving inventory control to arrive at cold shutdown.

3.3.3 Makeup Valve Bypass For Low Flow

The makeup control valve is subjected to excessive wear at the normal low flow rate. A design review is underway to determine if a bypass valve should be added.

3.3.4 RC Pump Seal Injection Upgrade

Modifications are being evaluated to enhance the RC pump seal injection reliability.

3.3.5 Increase HPI Flow

The makeup and purification system has been upgraded to provide more HPI flow to the reactor vessel core. The increase flow results in requiring that two existing HPI pumps provide water to the core under the single failure criteria. The MU/HPI pumps have been arranged to also consider maintenance requirements; therefore, three MU/HPI pumps are included in each of two trains which are supplied by separate emergency power.

3.3.6 Remote MU Pump Recirculation

A makeup pump recirculation will be provided to insure minimum pump flow requirements are met.

3.3.7. HPI Cross Connects

HPI cross connects are being evaluated to assure the identified minimum flow for HPI can reach the core. HPI cross connects will be added if they are required to assure the minimum flow is obtained with the upgraded MU&P system pump arrangement.

3.4 Decay Heat System

3.4.1 Dump to Sump

To prevent areas of high and/or low boric acid concentration in the core following a LOCA, a positive flow path through the core will be provided from the hot leg to the sump. This flow path will provide circulation through the core and prevent boron precipitation.

3.5 Core Flood System

3.5.1 Remote Defeat of Core Flooding Function

The ability will be provided to depressurize the core flooding tanks prior to depressurizing the RCS for safety grade cold shutdown.

3.6 Items Not Related To a Specific System

3.6.1 SBLOCA Mitigation

Plant specific analysis will be performed for North Anna Unit 3 using the improved analytical codes.

3.6.2 Natural Circulation and Single Loop Cooldown

Single loop and two-loop natural circulation analyses will be conducted to verify equipment changes and/or operator actions and procedure changes.

3.6.3 Containment Isolation Review

A review will be performed to look at the containment isolation design and design bases.

3.6.4 Overpressure Protection (OP)

Measures are being designed into the plant for North Anna Unit 3 to address the concerns of overpressure protection such as:

Low Temperature Overpressure Protection

Fail First Trip

Additional Pressurizer Safety Valve Capacity

3.6.5 Pressurizer Heater Qualification

Two groups of pressurizer heaters are being considered for safety grade application.

3.6.6 Incore Monitoring System (IMS)

Modifications to the IMS are being considered. The potential modifications would encompass the incore thermocouples and neutron flux distribution monitoring.

3.6.7 Neutron Flux Measurement

Modifications to the out-of-core neutron flux monitoring system are also being considered.

3.6.8 CRD Control System Upgrades

A ratchet trip prevention sub-system is being evaluated to be incorporated into the CRD Control System. Incorporation of CRD System improvements to reduce ratchet trips will reduce the number of transient operations to the system and improve plant availability by reducing the number of forced outages to remove or repair damaged CRD mechanisms.

3.6.9 Control and Data Acquisition Systems

The control and data acquisition systems are being evaluated for RG 1.75 and other licensing criteria not presently included in the hardware which has been built. New system descriptions and equipment specifications would be prepared and equipment supplied accordingly.

3.6.10 Statistical Core Design (SCD)

Under consideration for application to North Anna 3 is B&W's statistical Core Design Methodology which is an improved thermal-hydraulic core design/analysis technique that considers analysis uncertaintities by statistical combination rather than by compounding them as is done in traditional analysis techniques. The benefits of SCD are that the overall uncertainty penalty is reduced, thereby providing improved thermal margin, the core-wide protection for DNB-limited transients is better quantified, and the method of treatment of uncertainties provides a convenient means of addressing changes in uncertainty values or of incorporating additional parameters in the uncertainty allowance without requiring substantial analysis.

3.6.11 Power Upgrade

An upgrade of the North Anna 3 power level from 2631 to 2763 MWt is being evaluated.

3.6.12 RC Pump Trip

In selected RCS breaks the continued operation of the RC pumps could be undesirable.

Analyses will be performed to determine if an automatic RC pump trip is required for North Anna 3. With increased HPI flow, an automatic RC pump trip may not be required.

3.6.13 Inadequate Core Cooling Protection and Recovery

Analysis and procedures will be investigated for prevention and recovery from inadequate core cooling events.

3.6.14 Fuel Handling Equipment Upgrade

Several fuel handling upgrades are being considered.

3.6.15 Abnormal Transient Operating Guidelines (ATOG)

Symptom oriented emergency guidelines are being evaluated.