VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

RECEIVED
SEP 17 1981
U.S. NUCLEAR REGULATORY COMMISSION

R. H. LEASBURG
VICE PRESIDENT
NUCLEAR OPERATIONS

September 14, 1981

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
Attn: Mr. Steven A. Varga, Chief
Operating Reactor Branch No. 1
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Serial No.: 539
FR/RWC: plc
Docket Nos.: 50-280
50-281
License Nos.: DPR-32
DPR-37

Gentlemen:

AMENDMENT TO OPERATING LICENSES DPR-32 AND DPR-37
SURRY POWER STATION UNITS NO. 1 AND NO. 2
SUPPLEMENT TO PROPOSED TECHNICAL SPECIFICATIONS CHANGE

Pursuant to the provisions of Surry Units No. 1 and No. 2 Technical Specification 6.6.2.a(8), NRC Region II was notified on July 10, 1981 of an error in the currently applicable Loss of Normal Feedwater analysis. The error found was in the lack of representation of the reactor coolant pump (RCP) heat input after reactor trip in the Final Safety Analysis Report (FSAR) analysis. The impact of this error is to require more auxiliary feedwater (AFW) flow than can be provided by one AFW pump (350 gpm) to meet the analysis acceptance criteria. To determine an appropriate AFW flow which can satisfy the acceptance criteria when RCP pump heat input is continued after reactor trip, Vepco has performed licensing analyses using the RETRAN computer code. These analyses have been performed using the reactor system transient analysis methodology described in our topical report which was transmitted by letter from Mr. W. N. Thomas (Vepco) to Mr. H. R. Denton dated April 14, 1981 (Serial No. 215). Documentation of the new reference analysis to the level of detail consistent with the Surry FSAR is provided in Attachment 1. This documentation will be incorporated into the Surry Updated FSAR, which will be submitted to you no later than July 22, 1982. The analysis results indicate that an AFW flow of 500 gpm will meet the analysis criteria by preventing water relief from the pressurizer and ensuring long term decay heat removal. These results have been reviewed by both the Station Nuclear Safety and Operating Committee and the Safety Evaluation and Control staff.

In conjunction with this submittal, we would like you to proceed with your review of the license amendment applications discussed in our letter of April 20, 1981 (Serial No. 244). These license amendments include a proposed change to the Technical Specifications which would require three operable auxiliary feedwater pumps on each unit. Upon completion of your review and prior to issuance of the license amendments, please contact us in

A001
S 1/1

order to coordinate implementation of the license amendments with the status of modifications to the AFW system, which are currently scheduled for completion during the the upcoming Surry Unit No. 2 refueling. The administrative procedures provided in our LER 81-025/01X-1 as corrective actions will be followed in the interim to ensure that minimum auxiliary feedwater flow requirements are met.

If you require any additional information, please contact this office.

Very truly yours,

R. H. Leasburg

Attachment

1. Safety Evaluation

cc: Mr. James P. O'Reilly
Office of Inspection and Enforcement
Region II

COMMONWEALTH OF VIRGINIA)
)
CITY OF RICHMOND)

The foregoing document was acknowledged before me, in and for the City and Commonwealth aforesaid, today by R. H. Leasburg, who is Vice President-Nuclear Operations, of the Virginia Electric and Power Company. He is duly authorized to execute and file the foregoing document in behalf of that Company, and the statements in the document are true to the best of his knowledge and belief.

Acknowledged before me this 14th day of Sept., 19 81.

My Commission expires: 2-26, 19 85.

Ann C. McGee
Notary Public

(SEAL)

ATTACHMENT 1

SAFETY EVALUATION

LOSS OF NORMAL FEEDWATER

Identification of Causes and Accident Description

A loss of normal feedwater (from a pipe break, pump failures, valve malfunctions or loss of offsite AC power) results in a reduction in capability of the secondary system to remove the heat generated in the reactor core. If the reactor is not tripped during this accident, core damage would possibly occur from a loss of heat sink. If an alternative supply of feedwater is not supplied to the plant, residual heat following reactor trip would heat the primary system water to the point where water relief of the pressurizer occurs. Significant loss of water from the Reactor Coolant System (RCS) could conceivably lead to core damage. Since the plant is tripped well before the steam generator heat transfer capability is reduced, the primary system variables never approach a DNB condition.

The following provide the necessary protection against a loss of normal feedwater:

1. Reactor trip on low-low water level in any steam generator.

2. Reactor trip on high pressurizer level.

3. Reactor trip on high pressurizer pressure.

4. Two motor driven auxiliary feedwater pumps which are started on:

a. Low level in any steam generator
b. Any safety injection signal
c. Trip of all main feedwater pumps
d. Loss of offsite a-c power
e. Manual actuation

5. One turbine driven auxiliary feedwater pump is started on:

a. Low level in any two steam generators .
b. Manual actuation

The motor driven auxiliary feedwater pumps are supplied by the diesels if a loss of offsite power occurs and the turbine-driven pump utilizes steam from the secondary system. Both type pumps are designed to start within one minute even if a loss of offsite AC power occurs simultaneously with loss of normal feedwater. The turbine exhausts the secondary steam to the atmosphere. The auxiliary pumps take suction from the condensate storage tank for delivery to the steam generators.

The analysis shows that following a loss of normal feedwater, the Auxiliary Feedwater System is capable of removing the stored and residual heat thus preventing either overpressurization of the RCS or loss of water from the reactor core.

Analysis of Effects and Consequences

Method of Analysis

A detailed analysis using the RETRAN code has been performed in order to obtain the plant transient behavior following a loss of normal feedwater. The

simulation describes the plant neutron kinetics, RCS including natural circulation, pressurizer, steam generators and feedwater system. The digital program computes pertinent variables including the steam generator shell side inventory, pressurizer water level, and reactor coolant average temperature.

Assumptions:

1. The plant is initially operating at 102 percent of the ESF design rating.
2. A conservative core residual heat generation based upon long term operation at the initial power level preceding the trip.
3. Reactor trip occurs on low-low steam generator level.
4. Only two motor driven auxiliary feedwater pumps are available one minute after the accident.
5. Auxiliary feedwater is delivered to all steam generators.
6. Secondary system steam relief is achieved through the self-actuated safety valves. Note that steam relief will, in fact, be through the power-operated relief valves or condenser dump valves for most cases of loss of normal feedwater. However, for conservatism these have been assumed unavailable.
7. The initial reactor coolant average temperature is 4°F higher than the nominal value, and the initial pressurizer pressure is 30 psi higher than nominal.

The loss of normal feedwater analysis is performed to demonstrate the adequacy of the reactor protection and engineered safeguards systems (e.g., the Auxiliary Feedwater System) in removing long term decay heat and preventing excessive heatup of the RCS with possible resultant RCS overpressurization or loss of RCS water.

As such, the assumptions used in this analysis are designed to minimize the

energy removal capability of the system and to maximize the possibility of water relief from the coolant system by maximizing the coolant system expansion, as noted in the assumptions listed above.

For the loss of normal feedwater transient, the reactor coolant volumetric flow remains at its normal value and the reactor trips via the low-low steam generator level trip with no change in DNBR below the value at the start of the transient. The reactor coolant pumps may be tripped at some later time to reduce heat addition to the RCS; however, in this analysis they were conservatively assumed to be operating during the entire transient.

An additional assumption made for the loss of normal feedwater evaluation is that the pressurizer power-operated relief valves are assumed to function normally. If these valves were assumed not to function, the reactor coolant system pressure during the transient would rise to the actuation point of the pressurizer safety valves (2500 psia). The increased RCS pressure, however, results in less expansion of the coolant and hence more margin to the point where water relief from the pressurizer would occur. The balance of plant assumptions used in the analysis are listed in Table 1.

Results

Figures 1 through 4 show the significant plant parameter transients following a loss of normal feedwater. The calculated sequence of events for this accident is listed in Table 2.

Following the loss of main feedwater and prior to turbine trip, the water level in the steam generators will fall due to the continued steam flow to the turbine. Following the reactor trip on low-low steam generator water level and the subsequent turbine trip, steam flow through the safety valves continues to dissipate the stored and generated heat. One minute following the initiation of the low-low level trip, the auxiliary feedwater pumps are automatically started, reducing the rate of water level decrease.

The capacity of the auxiliary feedwater pumps is such that the water level in the steam generators does not recede below the lowest level at which sufficient heat transfer area is available to dissipate core residual heat without water relief from the pressurizer relief or safety valves. From Figure 2 it can be seen that at no time is there water relief from the pressurizer.

Conclusions

Results of the analysis show that a loss of normal feedwater does not adversely affect the core, the RCS, or the steam system since the auxiliary feedwater capacity is such that reactor coolant water is not relieved from the pressurizer relief or safety valves, and the water level in the steam generators remains adequate to remove decay heat from the core.

TABLE 1

BALANCE OF PLANT ASSUMPTIONS USED IN

LOSS OF NORMAL FEEDWATER ANALYSIS

Item	Value Used in Analysis	Criteria
Auxiliary Feedwater Delay Time	60 sec	At least two motor-driven auxiliary feedwater pumps will attain rated flow within 60 seconds after initiation of accident
Auxiliary Feedwater Temperature	120 °F	≤ 120 °F
Number of Auxiliary Feedwater Pumps	2	Minimum of 2 pumps must deliver to the steam generators
Auxiliary Feedwater Flow Rate	500 gpm	≥ 500 gpm

TABLE 2
TIME SEQUENCE OF EVENTS FOR
LOSS OF NORMAL FEEDWATER

Event	Time (sec.)
Analysis begins (steady state)	0
Main feedwater flow stops instantaneously	10
Low steam generator water level trip	71.8
Rods begin to drop	73.8
Peak water level in pressurizer occurs	77.0
Steam generators begin to receive auxiliary feed from two motor-driven auxiliary feedwater pumps	131.8
Auxiliary feedwater addition matches secondary boiloff required to remove decay heat and pump heat. Steam generator inventory begins to increase.	~1200

FIGURE 1

PRESSURIZER PRESSURE

LOSS OF NORMAL FEEDWATER TRANSIENT

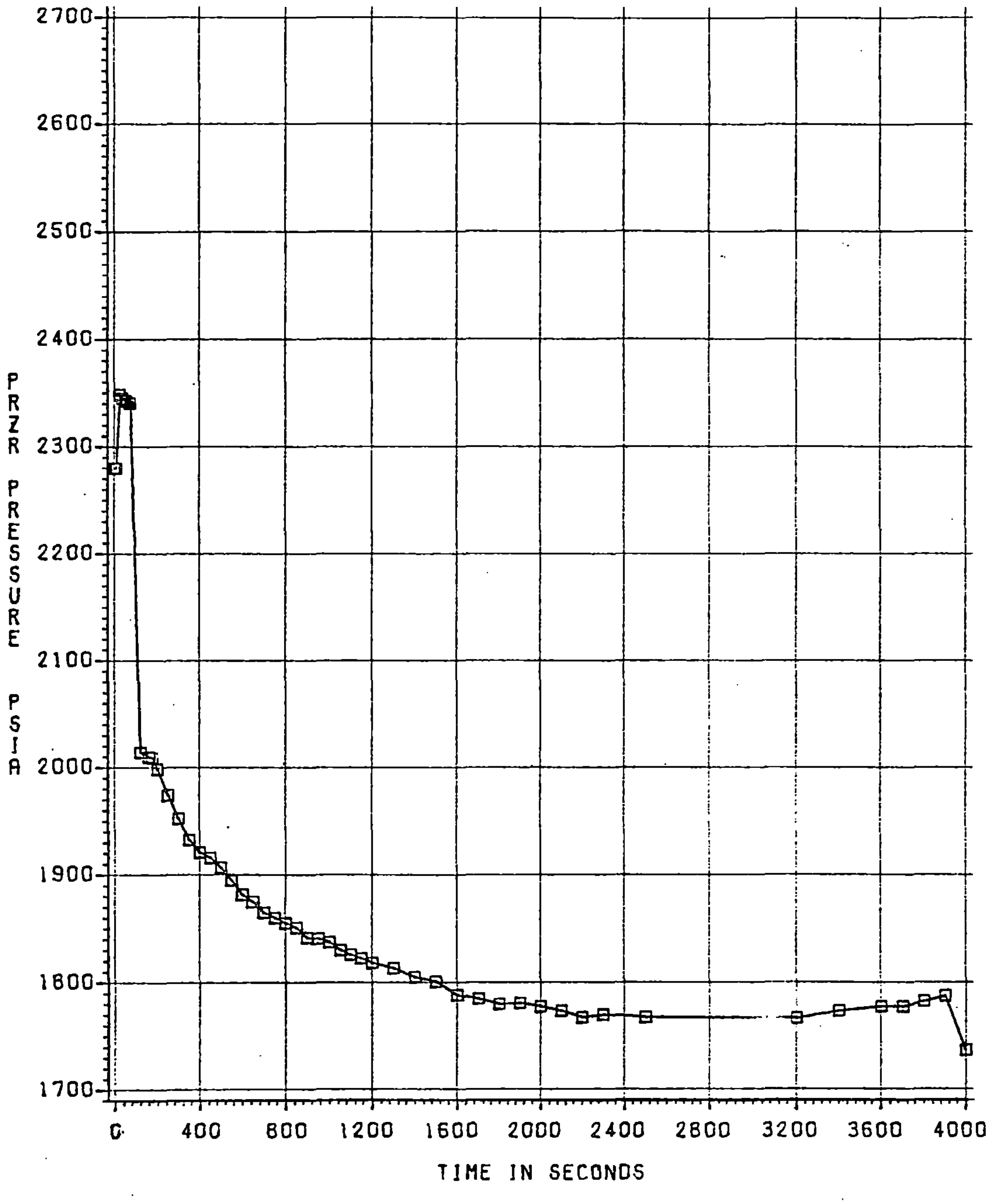

FIGURE 2

PRESSURIZER WATER VOLUME

LOSS OF NORMAL FEEDWATER TRANSIENT

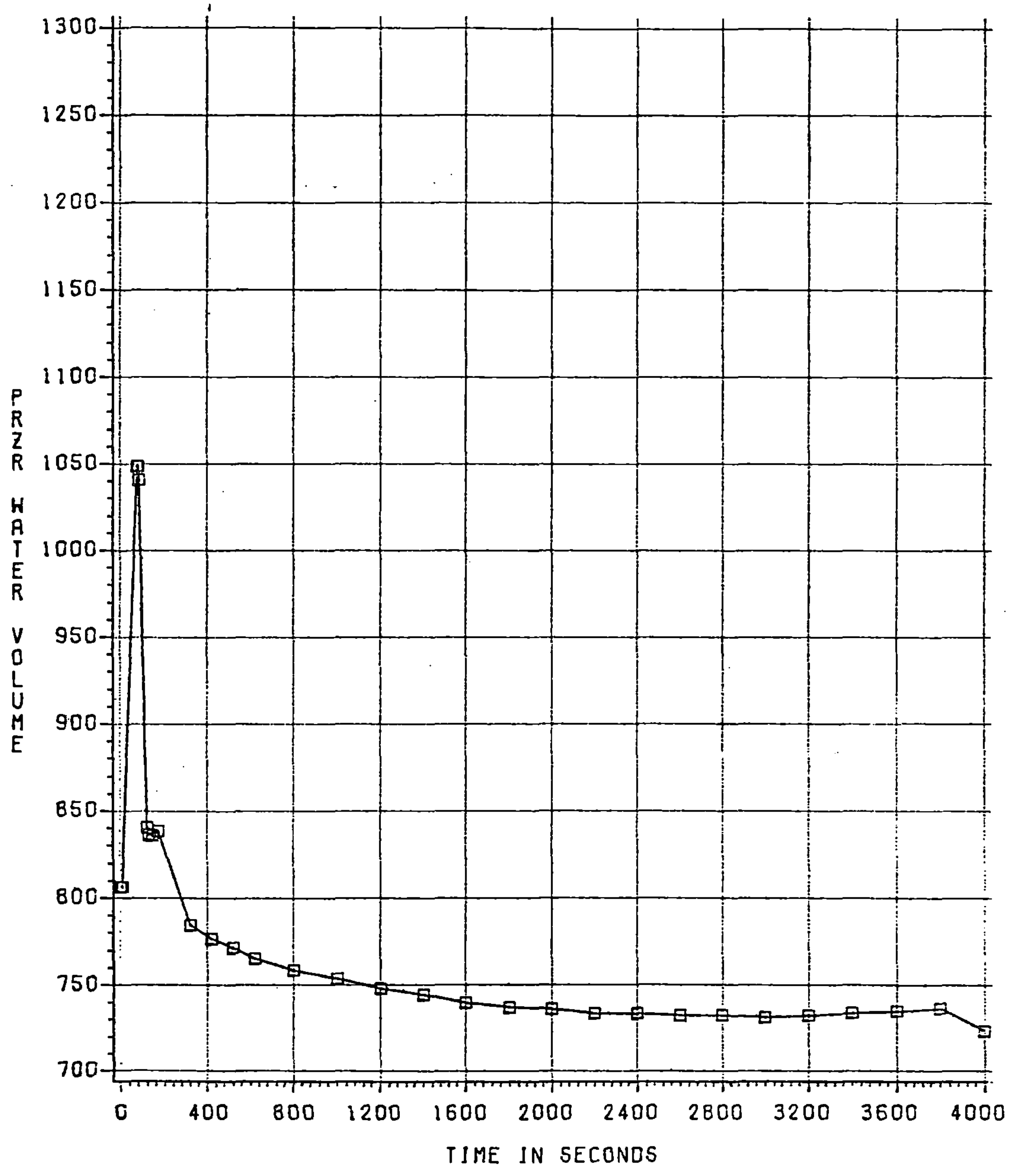

FIGURE 3

LOOP TEMPERATURES

LOSS OF NORMAL FEEDWATER TRANSIENT

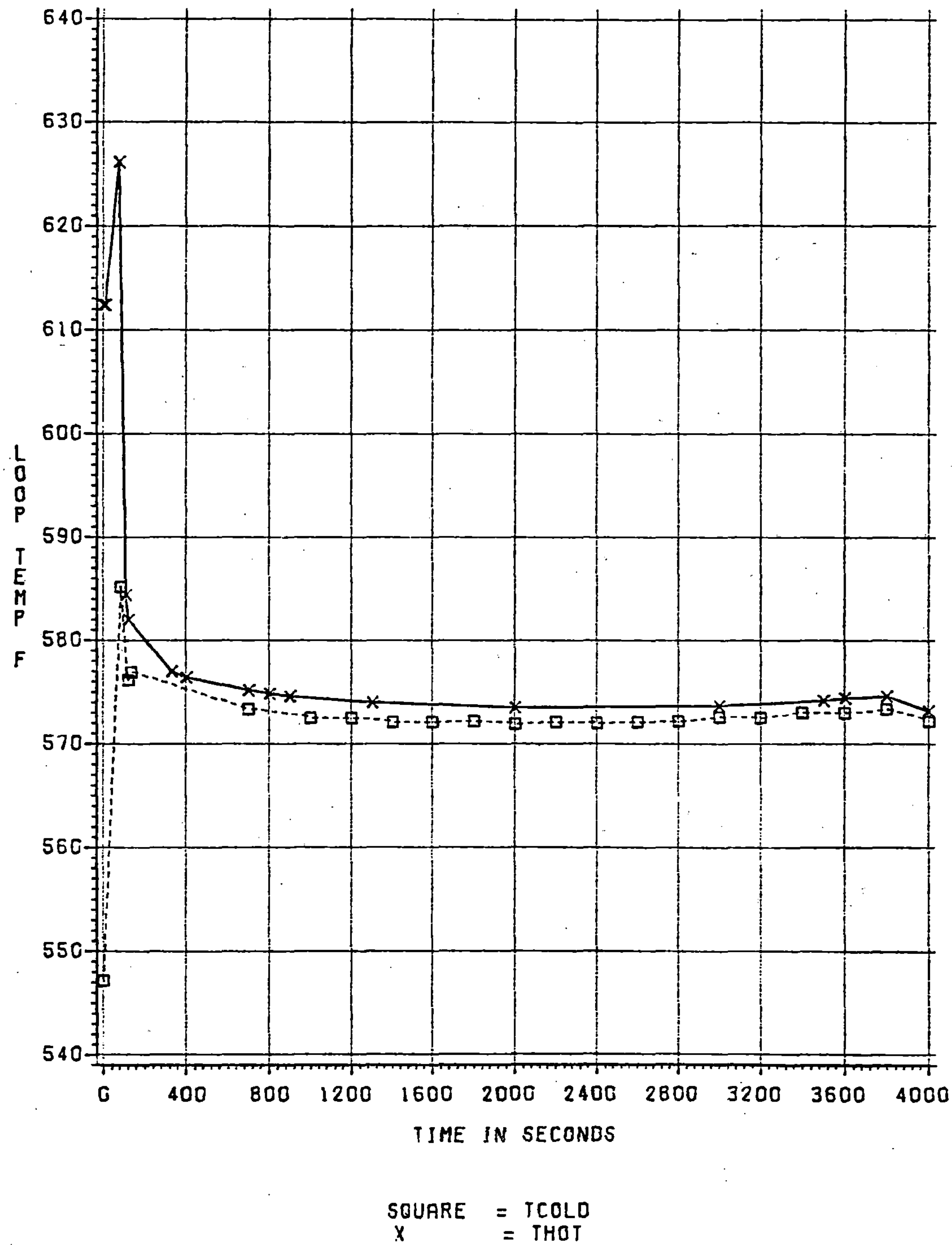

FIGURE 4

STEAM GENERATOR PRESSURE

LOSS OF NORMAL FEEDWATER TRANSIENT