Public Service
Electric and Gas
Company

Corbin A. McNeill, Jr.
Senior Vice President -
Nuclear

Public Service Electric and Gas Company P.O. Box 236, Hancocks Bridge, NJ 08038 609 339-4800

July 2, 1987

NLR-N87124
LCR 87-09

United States Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

Gentlemen:

REQUEST FOR LICENSE AMENDMENT
FACILITY OPERATING LICENSE DPR-70 AND DPR-75
SALEM GENERATING STATION - UNIT NOS. 1 AND 2

In accordance with the Atomic Energy Act of 1954, as amended, and the regulations thereunder, we hereby transmit our request for amendment and our analyses of the changes to Facility Operating Licenses DPR-70 and DPR-75 for the Salem Generating Station Unit Nos. 1 and 2, respectively.

The amendment request consists of changing the reactor trip block with a turbine trip from the P-7 (11% power) permissive up to the P-9 (50% power) permissive. This will allow the Salem units to sustain a turbine trip without causing a reactor trip up to 50% of rated thermal power. This change allows operation similar to that licensed on numerous other Westinghouse plants. These changes can be incorporated during any plant outage and currently scheduled for the cycle 4 outage for Unit 2 (April 1988) and for P-8 permissive modifications during the cycle 7 outage (September 1987) and P-9 permissive modifications during the cycle 8 outage (April 1989) for Unit 1. Therefore, approval by September 1987 is requested.

Enclosed is a check in the amount of $150.00 as required by 10CFR 170.21.

Pursuant to the requirements of 10CFR50.91, a copy of this request for amendment has been sent to the State of New Jersey as indicated below. This submittal consists of one (1) signed original and thirty-seven (37) copies.

A001 \1037 w/check $150 #1177

Should there be any questions regarding this matter, please feel free to contact us.

Sincerely,

Enclosures

C Mr. D. C. Fischer
USNRC Licensing Project Manager

Mr. T. J. Kenny
USNRC Senior Resident Inspector

Mr. W. T. Russell, Administrator
USNRC Region 1

Mr. D. M. Scott, Chief
Bureau of Nuclear Engineering
Department of Environmental Protection
380 Scotch Road
Trenton, NJ 08628

Ref: LCR 87-09

STATE OF NEW JERSEY)
) SS.
COUNTY OF SALEM)

Corbin A. McNeill, Jr., being duly sworn according to law deposes and says:

I am Senior Vice President of Public Service Electric and Gas Company, and as such, I find the matters set forth in our letter dated July 2, 1987 , concerning Facility Operating Licenses DPR-70 and DPR-75 for Salem Generating Station, is true to the best of my knowledge, information and belief.

Subscribed and Sworn to before me
this 2nd day of July , 1987

Notary Public of New Jersey

LARAINE Y. BEARD
Notary Public of New Jersey
My Commission Expires May 1, 1991

My Commission expires on ______________________

ATTACHMENT 1

PROPOSED CHANGES TO TECHNICAL SPECIFICATIONS LCR 87-09
SALEM GENERATING STATION, UNIT NOS. 1 & 2
FACILITY OPERATING LICENSE DPR-70 & DPR-75
DOCKET NOS. 50-272 & 50-311

I. Description of the Change

Revise Technical Specification Bases Section 2.2.1, Turbine Trip, and Table 3.3-1, Reactor Trip System Instrumentation, to indicate that the P-9, rather that the P-7, Permissive Setpoint will defeat the automatic block of a reactor trip on a turbine trip when 2 of 4 Power Range Neutron Flux Channels are greater than or equal to 50%, rather than 11%, of RATED THERMAL POWER. This change will permit continued reactor operation following turbine trips, provided that reactor power is no greater than 50% of RATED THERMAL POWER. This will enable the plant to either restart the turbine for trips which are readily correctable or to commence an orderly reactor shutdown.

II. Reason for the Change

The basic design philosophy for Westinghouse Pressurized Water Reactors (PWRs) has included a reactor trip following a turbine trip whenever the plant is operating above some nominal power level. This reactor trip is not required but rather exists as an anticipatory trip to prevent a trip associated with excessive reactor coolant temperature and pressure rises and to enhance the overall reliability of the Reactor Protection System (RPS). Currently, the design of Salem Generating Station, both Unit Nos. 1 and 2, includes a reactor trip when reactor power is above 11% of RATED THERMAL POWER if a major load loss results from a loss of external electrical load or a turbine trip. For either case offsite power is available for the continued operation of vital plant components.

The SGS Turbine Bypass System (see Updated Final Safety Analysis Report (UFSAR) Section 10.4.4.1) provides the capability to dump up to 40% of full load steam flow directly to the condenser which enables the plant to accept a step load decrease of 50% of full load from full load without a reactor trip (the remaining 10% of the 50% is an inherent capability of the Nuclear Steam Supply System (NSSS) to accept a 10% step load change). Therefore, to trip the plant when operating below 50% of RATED THERMAL POWER upon an electrical loss or turbine trip is unnecessary if the cause of the turbine trip is readily correctable. Hence, this License Change Request (LCR) has been developed in order to increase plant availability by significantly reducing the down time required to restart the plant after an unnecessary plant trip, i.e. currently reactor trips are required for turbine trips when operating at greater than 11% of RATED THERMAL POWER but are only necessary when operating at greater than 50% of RATED THERMAL POWER.

PSE&G has obtained only enough hardware in order to install the P-9 setpoint on SGS Unit No. 2. Hence, SGS Unit No. 1 will utilize the P-8 setpoint (greater than or equal to 36% of RATED THERMAL POWER) until such time as the P-9 hardware is available and installed. In order to eliminate the need for a second LCR for SGS Unit No. 1 (i.e. one for P-8 and another for P-9), the justification and significant hazards analysis provided for this LCR have been developed assuming the P-9 setpoints will be installed and the LCR discusses the use of a setpoint at 50% of RATED THERMAL POWER. However, Subparagraph IV.B.3 below discusses the use of an interim setpoint at 36% of RATED THERMAL POWER (P-8) and the SGS Unit No. 1 Technical Specification revisions contained in Attachment 3 contain provisions for use of the P-8 setpoint until such time as the P-9 setpoint is installed and available.

III. Justification for the Change

A summary discussion is warranted regarding the design for tripping the plant in the event of a loss of external electrical load or a turbine trip when operating above some nominal power level. Currently this level is set at 11% of RATED THERMAL POWER and is known as the P-7 permissive setpoint. The basis for this setpoint is derived from the standard Westinghouse Technical Specifications.

For the loss of external electrical load, the turbine is automatically tripped when both generator output breakers trip open. UFSAR Section 10.2.2.4 identifies a variety of conditions under which the turbine is tripped due to station events. Once the turbine is tripped and the P-7 permissive setpoint is satisfied (i.e. at greater than 11% of RATED THERMAL POWER), the reactor would be tripped on signals that the turbine auto stop emergency trip fluid pressure fell below 45 psig or the four turbine stop valves were less than 85% full open - the turbine stop valves close rapidly (typically within 0.1 seconds) on loss of trip-fluid pressure. Upon closure of the four stop valves, steam flow to the turbine is stopped which results in an almost immediate rise in secondary system temperature and pressure. This increase results in a reduced heat transfer rate in the steam generator, causes the reactor coolant to expand, creates a pressurizer insurge and causes the reactor coolant system pressure to rise. These conditions would create a situation in which the reactor is tripped on high pressurizer pressure, overtemperature delta-T, or low-low steam generator level, if the turbine trip had not already tripped the plant.

If the automatic steam dump system is operating properly, up to 40% of full load steam flow (50% of RATED THERMAL POWER) can be bypassed to the condensor without a secondary-to-primary system transient causing a reactor scram. The secondary-to-primary system scenerio described above would not occur even if the currently design P-7 setpoint failed to function. Hence at power levels above 11% but below 50% of RATED THERMAL POWER, the anticipatory reactor trip upon turbine trip is unnecessary.

IV. Significant Hazards Consideration

In order to develop the proposed change fully, PSE&G has evaluated a variety of plant scenerios and previously completed accident analyses to assess the impact of increasing the turbine-reactor trip setpoint to 50% from 11% of RATED THERMAL POWER. The scenerios include:

i. UFSAR Section 15.2.5 discusses the plant response to a partial loss of forced reactor coolant flow. The accident analysis was re-evaluated in light of the revised setpoint, including the impact on the 30 second turbine-generator motoring feature. The results are presented in Item A.1 below.

ii. UFSAR Section 15.2.7 discusses the plant response to a loss of external load and/or turbine trip. This accident analysis was re-evaluated in light of the revised setpoint and the results are presented in Item A.2 below.

iii. UFSAR Section 15.3.4 discusses the plant response to a complete loss of forced reactor coolant flow. This accident analysis was re-evaluated in light of the revised setpoint and the results are presented in Item A.3 below.

iv. As discussed in Paragraph III above, the operation of the steam dump system (i.e. the Turbine Bypass System) was assumed in order for the plant to continue to operate following a turbine trip below 50% of RATED THERMAL POWER. Therefore, the plant response to a failure of the Turbine Bypass System at between 11% and 50% of RATED THERMAL POWER was evaluated and the results are presented in Item B.1 below.

v. A study of the potential for increased pressurizer PORV opening resulting from a turbine trip without a reactor trip below 50% of RATED THERMAL POWER was completed by Westinghouse for PSE&G. The results of this transient analysis are summarized in Item B.2 below and further details are provided in Attachment 2.

vi. In the case of SGS Unit No. 1, the use of the P-9 setpoint is restricted until the receipt and installation of hardware required to actually change the setpoint (see the discussion contained in Paragraph II above). Until such time, Unit No. 1 intends to operate with the P-8 setpoint as the turbine-reactor trip permissive. Therefore, the plant response to the use of a 36% of RATED THERMAL POWER setpoint was evaluated and the results are presented in Item B.3 below.

The proposed changes to the SGS Technical Specifications:

A. Do not involve a significant increase in the probability or consequences of an accident previously evaluated.

1. UFSAR Section 15.2.5 contains an analysis of the Condition II Partial Loss of Forced Reactor Coolant Flow. The accident scenerio evaluates a fault in the power supply to a reactor coolant pump which can result in a partial loss of coolant flow. If the reactor is at power at the time of the accident, the immediate effect of loss of coolant flow is a rapid increase in the reactor coolant temperature. The accident analysis assumed that the low reactor coolant flow signal is available to trip the reactor. Above 36% of RATED THERMAL POWER (i.e. the P-8 setpoint), low flow in any one of the four loops will actuate a reactor trip, while between 11% and 36% of RATED THERMAL POWER (i.e. between P-7 and P-8) low flow in any two of four loops will actuate a reactor trip. A reactor trip signal from the reactor coolant pump circuit breaker open position is provided as an anticipatory trip signal which serves to backup the low flow signals.

Normal power for each of the reactor coolant pumps is supplied through separate busses from a transformer connected to the generator. Following any turbine trip where there are no electrical faults which require tripping the generator from the network, the generator remains connected to the network for approximately 30 seconds. The reactor coolant pumps remain connected to the generator thus ensuring full flow for approximately 30 seconds after the reactor trip before any transfer is made. When the generator trip occurs, the busses are automatically transferred to a transformer supplied from external power lines. This feature is known as turbine-generator motoring and is provided so that full reactor coolant flow is maintained to remove reactor core heat during Condition II overpower transients (UFSAR Section 15.2.5) and to prevent any pump overspeed conditions (see UFSAR Section 5.5.1.3.9).

If the entire 30 second time delay fails or if an electrical fault exists such that the generator is immediately tripped from the network, a fast bus transfer to offsite power will be initiated by the generator trip signal. This transfer ensures reactor coolant flow by transferring the reactor coolant pumps within six to ten cycles. The RPS initiates a reactor trip as protection for an overpower event while the turbine protection system has a mechanical overspeed trip at 110% of turbine speed and the turbine control system and intercept valves limit the overspeed to 120% of turbine speed to prevent the consequences of a pump overspeed condition.

Changing the turbine-reactor trip setpoint from 11% to 50% of RATED THERMAL POWER will not affect the low reactor coolant loop flow or reactor coolant pump circuit breaker open position signals. The change removes the defeat of the automatic block of reactor trip upon turbine trip from P-7, which is an independent trip signal from the two signals provided for protection against a partial loss of coolant flow. Above 50% of RATED THERMAL POWER, the 30 second turbine-generator motoring feature is still available as the P-9 setpoint only serves to trip the reactor upon a turbine trip. The 30 second time delay associated with the generator decoupling from the network is an independent feature from the turbine trip signals and even if this feature fails upon turbine trip, sufficient RPS signals are available as described above to mitigate an overpower transient. As a result, the UFSAR Section 15.2.5 accident analysis bounds the consequences of a turbine trip event below 50% of RATED THERMAL POWER with or without turbine-generator motoring and hence the proposed changes do not increase the possibility or consequences of this previously evaluated accident.

A discussion of a failure of the transfer following the turbine-generator motoring 30 second time limit is discussed in Items A.2 and A.3 below and the consequences of a failure of the steam dump is discussed in Item B.1 below.

2. UFSAR Section 15.2.7 contains an analysis of the Condition II Loss of External Electrical Load and/or Turbine Trip. The description of the accident sequence identifies that the reactor would be tripped directly from a signal derived from the turbine auto stop oil pressure and/or the turbine stop valves unless below 11% of RATED THERMAL POWER, that power level associated with the P-7 permissive setpoint. The analysis however contains several conservative assumptions including: (1) a complete loss of steam load at 102% of full power, (2) no direct reactor trip upon turbine trip, and (3) no credit for steam dump or steam generator power operated relief valves. Four accident evaluations were completed using the LOFTRAN digital computer program assuming beginning of life

conditions for minimum moderator reactivity feedback and end of life conditions for maximum moderator reactivity feedback both of which were completed assuming credit for the effect of pressurizer spray and power operated relief valves (PORVs) as well as assuming no credit for pressurizer spray and PORVs. These four cases have been evaluated using the P-9 setpoint (nominal 50% power).

i. The analyses performed without pressure control, both minimum and maximum reactivity feedback cases, indicate that a reactor trip on high pressurizer pressure will occur within 6-7 seconds if the event is initiated from full power, and within about 12-17 seconds if the event is started from 50% power. The power and temperature conditions which exist at full power are more limiting than at any partial power with respect to the minimum DNBR reached during the transient (in fact, the DNBR will increase throughout the transient). The pressurizer safety valve setpoint will be reached for both the full and partial power cases which assume no pressure control with the pressurizer PORVs or sprays. However, the event initiated from partial power will turn around faster due to the lower initial power and temperatures in addition to the lower amount of stored energy in the fuel. For both the full power and partial power evaluations, the reactor trip occurs long before the fast bus transfer is attempted, thus the loss of flow which may occur due to the fast bus transfer failure after 30 seconds of turbine-generator motoring will have no effect on the transient for either the full or partial power cases.

ii. The analysis performed at partial power (50%) with pressure control from the pressurizer PORVS and sprays, for both the minimum and maximum moderator reactivity feedback cases, indicate that the reactor may not trip until an undervoltage or low flow setpoint is reached after failure of the fast bus transfer 30 seconds into the event. The power, temperature and pressure conditions which exist at the time of the loss of flow for the partial power cases are much less severe with respect to minimum DNBR than those conditions which exist for the complete loss of flow event (UFSAR Section 15.3.4). The largest benefit is due to the lower initial power level at the time of the loss of flow.

For the minimum feedback case, the reactor will essentially remain at the 50% power level until the reactor trip occurs on the reactor coolant pump undervoltage or low flow signal. The reactor coolant average temperature increases for the partial power loss of load event; however, the RCS average temperature will not significantly increase above the nominal full power RCS average temperature and thus will not offset the DNB benefit from the large difference in power level. The power transient for the maximum reactivity feedback case will steadily decrease from the initial power level due to the heatup and moderator feedback effect. The RCS average temperature for this case will increase, but not above the nominal full power RCS average temperature, so the resultant minimum DNBR will be greater than the minimum DNBR for the loss of flow event (UFSAR Section 15.3.4). Therefore, the FSAR full power complete loss of flow event bounds the partial power cases with respect to the minimum DNBR reached during the transient. With the PORVS and sprays available for pressure control, both the full and partial power cases will show a pressure increase on the primary side to the PORV setpoint. After reactor trip, the pressure will decrease throughout the remainder of the transient. The cases without pressure control are always more limiting with respect to peak pressures and, as stated above, the UFSAR full power loss of load event will bound any partial power event with respect to peak pressure.

Finally, the proposed change would increase the turbine trip to 50% of RATED THERMAL POWER and hence the reactor would not be tripped above the 11% RATED THERMAL POWER level identified in the accident description. However, since the LOFTRAN computer program did not take credit for the direct turbine-reactor trip, whether such a trip takes place or not does not affect the results of this accident analysis. Hence changing the turbine-reactor trip setpoint from 11% to 50% of RATED THERMAL POWER has no bearing on the results of the loss of external electrical load and/or turbine trip. As a result, the UFSAR Section 15.2.7 accident analysis bounds the consequences of a turbine trip event below 50% of RATED THERMAL POWER with or without a subsequent reactor trip and hence the proposed changes do not increase the probability or consequences of this previously evaluated accident.

3. UFSAR Section 15.3.4 contains an analysis of the Condition III Complete Loss of Forced Reactor Coolant Flow. This accident sequence evaluates the effects of a complete loss of forced reactor coolant flow from a loss of electrical power supply to the reactor coolant pumps. If the reactor is at power at the time of the accident and a failure of the network bus transfer occurs, the immediate effect is a more rapid increase in the coolant temperature compared to the increased coolant temperature as a result of the turbine trip by itself. The analysis assumed that the following RPS signals are available to trip the reactor: (1) undervoltage or underfrequency on the reactor coolant pump power supply busses, (2) low reactor coolant loop flow, or (3) reactor coolant pump circuit breaker opening. The reactor trip on reactor coolant pump undervoltage is provided to protect against a station blackout event and is blocked below 11% RATED THERMAL POWER (i.e. the P-7 setpoint). The reactor trip on reactor coolant pump underfrequency is provided to protect against

major power grid frequency disturbances by disengaging the pumps so that the pumps' kinetic energy is available for full coastdown. The reactor trip on low reactor coolant flow and reactor coolant pump circuit breaker opening were discussed in Item A.1 above.

Changing the turbine-reactor trip setpoint from 11% to 50% of RATED THERMAL POWER will not affect the RPS signals provided for protection against a complete loss of forced reactor coolant flow. The change removes the defeat of the automatic block of reactor trip upon turbine trip from the P-7 signal and adds the defeat to the P-9 permissive signal. This is an independent trip signal from the three signals identified above which remain unaffected. Between 11% and 50% of RATED THERMAL POWER the three signals identified above will still be available to mitigate the consequences of a complete loss of forced reactor coolant since the reactor coolant pumps will still be operating. Above 50% of RATED THERMAL POWER the three signals will function as they do currently for P-7. The installation of the P-9 setpoint in no way alters or affects the capability of the RPS to perform its intended function. As a result, the UFSAR Section 15.3.4 accident analysis bounds the consequences of a turbine trip event below 50% of RATED THERMAL POWER with or without a subsequent reactor trip and hence the proposed changes do not increase the probability or consequences of this previously evaluated accident.

B. Do not create the possibility for a new or different kind of accident than any previously evaluated.

1. Inherent within the proposed change is the capability of the Turbine Control System to handle a steam dump to the condensor at or below 50% of RATED THERMAL POWER. UFSAR Section 10.4.4.1 discusses the design of the steam dump control system and discusses the consequences should the system fail to operate both above and below 50% of RATED THERMAL POWER.

In the event of a loss of load or turbine trip at or below 50% RATED THERMAL POWER, and should the steam dump valves fail (or the condenser not be available as a heat sink), the steam generator pressure and primary system temperature will rapidly increase. The steam generator safety valves are sized to remove the steam flow at 105 percent of steam flow at rated power, well within the 110 percent of steam system design pressure. With the turbine condenser not available, steam will be dumped to the atmosphere and main feedwater flow will be lost. In this situation the feedwater flow will be maintained by the Auxiliary Feedwater System thereby insuring adequate residual and decay heat removal capability. When the steam generator safety valves are reseated, the pressurizer power operated relief valves (PORVs) will operate to remove residual heat and control steam pressure. The pressurizer PORVs are sized to relieve sufficient steam to maintain the reactor coolant system pressure within 110 percent of the reactor coolant system design pressure.

In the event of a loss of load or turbine trip above 50% of RATED THERMAL POWER or should it be necessary to close the Main Steam Stop Valves (MSSVs) under full load, safety relief valve capacity equal to 100% of full load flow is provided on the piping just upstream of the MSSVs. The capacity is provided by five self-actuated safety valves on each main steam line, with setpoints ranging from 1070 to 1125 psig, which vent via umbrella vents to atmosphere through the roof of the penetration area. Additionally, a power operated steam relief valve is provided on each main steam line upstream of the MSSVs, total capacity for all four valves is 10% of full load. These valves have remotely variable pressure setpoints and can be used to bleed off reactor decay heat.

As discussed above, the operation of the steam dump system is not the only means to control secondary system pressure following a load loss or turbine trip. Therefore, credit can be taken for the function of this system, and hence, it can be concluded that a reactor trip below 50% of RATED THERMAL POWER is not required following a loss of load or turbine trip from a pressure standpoint. Additionally, increasing the turbine-reactor trip setpoint from 11% to 50% of RATED THERMAL POWER does not create the possibility of a new or different accident in terms of the capability of the secondary system to handle full or partial load pressures in the event the turbine control system fails to automatically dump steam to the condenser.

2. Westinghouse has completed a study of the potential for increased pressurizer PORV opening from a turbine trip without a reactor trip at 50% of RATED THERMAL POWER and has concluded that (i) a turbine trip below 50% of RATED THERMAL POWER will not result in opening the pressurizer power operated relief valves, and (ii) that even considering the scenerio along with degraded control system performance (i.e. steam dump system, pressurizer spray system or rod control system failure), the pressurizer power operated relief valves will not open. The results of this study are presented in further detail in Attachment 2.

3. As discussed in Paragraph II above, the use of the P-9 setpoint for SGS Unit No. 1 is restricted until receipt and installation of hardware required to actually change the setpoint (currently PSE&G has only enough material onsite to install the modification for SGS Unit No. 1). As a result, PSE&G is proposing the use of the P-8 setpoint (i.e. at or below 36% of RATED THERMAL POWER) in the interim between NRC issuance of this amendment

request and the subsequent receipt and installation of the P-9 hardware. As discussed within this submittal, the proposed change does not affect the P-7 setpoint in any manner other than to remove the turbine-reactor trip permissive. Similarly, the addition of this permissive to the P-8 setpoint does not affect any other functions the P-8 setpoint currently performs. Additionally, the P-8 setpoint (less than or equal to 36% of RATED THERMAL POWER) is less than the P-9 setpoint (less than or equal to 50% of RATED THERMAL POWER) and hence, the accident analyses which bound the proposed change for P-9 also bound the proposed temporary change for P-8. Therefore, the use of a P-8 setpoint prior to the P-9 setpoint on SGS Unit No. 1 will not create the potential for any new or different kind of accident than previously evaluated.

C. Do not significantly reduce the margin of safety for any Technical Specification.

The proposed change does increase the turbine-reactor trip to 50% from 11% of RATED THERMAL POWER; however, this increase is not a significant change in the Technical Specification margins of safety. This conclusion can be reached because of the inherent design feature of SGS, namely that the turbine control system is already designed such that a 50% steam dump is within the operating limits of the station. Hence, the design of the plant is not changing, only the current Technical Specification governing turbine-reactor trips. In addition, the proposed change conforms to Example 6 of 48FR14870 in that the change is clearly within all acceptable criteria with respect to the system.

From the discussions provided above, PSE&G has concluded that the proposed change to the Technical Specifications does not involve a significant hazards consideration.

A STUDY OF THE POTENTIAL FOR INCREASED PRESSURIZER PORV OPENING RESULTING FROM TURBINE TRIP WITHOUT REACTOR TRIP BELOW 50% POWER TRANSIENT (P-9 SETPOINT STUDY)

I. INTRODUCTION

The Salem Units are designed with 50% load rejection capability. As a result of this capability, the Westinghouse design criterion is that load rejections up to 50% should not require a reactor trip if all other control systems function properly. Therefore, Westinghouse has proposed to implement an interlock system that would eliminate direct reactor trips on turbine trips below 50% power, thereby decreasing unnecessary challenges to the reactor protection system and increasing plant availability. The NRC has expressed concerns regarding the potential increase in probability of a stuck-open pressurizer PORV following the implementation of deletion of reactor trip on turbine trip below 50% power. The NRC position is addressed in NUREG-0737, Item II-K.3.10. The information contained herein presents a best estimate analytical study to show that no additional pressurizer PORV challenges are expected due to implementation of an interlock system that would eliminate direct reactor trips on turbine trips below 50% power for Salem Units.

II. SYSTEM TRANSIENT ANALYSIS

II.1 Description of the Analysis:

A best estimate analytical study was performed to determine the transient plant response to a turbine trip without a reactor trip from 50% power. The analysis was performed using the LOFTRAN computer code[1] model of the Salem Units. This computer model simulates the overall thermal/hydraulic/nuclear response of the NSSS as well as the various control and protection systems. Since the object of this study was primarily to determine the peak in pressurizer pressure following the initiation of the transient, assumptions

(1) LOFTRAN Code Description, WCAP-7878, Rev. 0 - Rev. 3.

were made that would contribute to a conservatively high prediction of pressurizer pressure. These assumptions were the following.

1. Beginning-of-Life (BOL) reactivity parameters were used since this gives the minimum moderator feedback, and consequently, the minimum decrease in nuclear power as a result of the initial increase in primary coolant temperature during the transient.

2. Transients were initiated from 52% power (2% calorimetric error in adverse direction) since 50% power is the maximum proposed value for the P-9 permissive setpoint that would permit a turbine trip without actuating a direct reactor trip. Transients initiated from a lower power level would be less severe with respect to predicting the peak in pressurizer pressure. This is true since the peak in pressurizer pressure is directly related to the amount of energy that must be stored in the primary system during the mismatch between core power production and secondary system load. A turbine trip from a lower initial power level simply results in a smaller power mismatch and this results in a smaller peak in pressurizer pressure. In the limit, the initial power level of the transient would be reduced to 10% power which is currently the power level at which a turbine trip without a reactor trip is permitted.

3. The pressurizer model in LOFTRAN is conservative with respect to over predicting peaks in pressurizer pressure. This is because the pressurizer pressure calculation model in LOFTRAN is isentropic. Comparison studies[2] have shown that such models (isentropic model) overpredict pressurizer pressure during pressure-increase transients.

---

(2) Baron, R. C., "Digital Model Simulation of a Nuclear Pressurizer," Nuclear Science and Engineering 52, 283-291 (1973).

II.2 Analysis Results:

The expected system response to a turbine trip without a reactor trip from 50% power is shown in figures 1 through 6. For normal plant operation with all normal control systems assumed operational, the pressurizer pressure does not reach the point of pressurizer PORV activation (PORV setpoint for the Salem Units is 2350 psia). Results also indicated that steam generator PORV does not open during this transient. Note that in figures 1 through 6, transient initiates after 10 seconds of steady state.

III. FAILURE MODES ANALYSIS

For normal plant operation with all normal control systems assumed operational, this transient does not result in opening the pressurizer PORVs. However, the NRC has expressed a concern that the implementation of a turbine trip without a reactor trip below 50% power permissive should not result in increased challenges to the pressurizer PORVs even in the event of "degraded" control system performance. A sensitivity study was therefore performed in which certain failures were assumed to occur in the control systems that influence the course of this transient in order to determine their effect on the potential for pressurizer PORV challenges.

There are three main control systems that act during this transient: the steam dump system, the pressurizer spray system and the rod control system. The steam dump system consists of twelve valves which are arranged into four banks, three valves per bank. A single credible failure was assumed to be the failure of a bank of steam dump valves to open on demand following the turbine trip. In the pressurizer spray system, the types of failure assumed were either a reduction in spray flow capacity (due, for example, to a sticking spray control valve) or a complete failure to receive any spray flow (due, for example, to a failure of the control actuating signal). The failure that was assumed in the rod control system was the failure of the power mismatch channel. The purpose of the power mismatch channel signal is to provide a fast feed-forward signal to the rod control system during a rapid change in turbine load. If this signal is not present, then the rods are controlled only by the Tavg error signal which has a much slower response and thus it takes longer time to begin driving the rods into the core following the turbine trip.

III.1 Failure Mode Analysis Results:

The results of the failure mode sensitivity study showed that for normal system operation and for any single failure that was considered, both pressurizer and steam generator PORVs did not open. In fact, it takes a combination of multiple control system failures to result in pressurizer or the steam generator PORVs opening during the transient.

IV. CONCLUSIONS

Based on the best estimate analysis results following conclusions are made:

1. For normal plant operation with all normal control systems assumed operational, the implementation of a system (P-9 permissive) that permits a turbine trip without actuating a direct reactor trip below 50% power will not result in opening the pressurizer power-operated relief valves.

2. For any single failure in the control system that was considered in the analysis, the implementation of a system (P-9 permissive) that permits a turbine trip without a direct reactor trip below 50% power will not result in opening the pressurizer power-operated relief valves. It was found that, it takes a combination of multiple control system failures to result in pressurizer power-operated relief valves opening during the transient.