UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
SOUTH CAROLINA ELECTRIC &	)	
GAS COMPANY	)	Docket No. 50-395
	)	
(Virgil C. Summer Nuclear	)	
Station, Unit 1)	)	

AFFIDAVIT OF ROBERT W. STEITLER ON
ANTICIPATED TRANSIENTS WITHOUT SCRAM

My name is Robert W. Steitler and my qualifications can be found in Attachment A. The purpose of this review is to explain what an ATWS event is, to review the analyses for Westinghouse pressurized water reactors like Summer, which have superior inherent ATWS mitigation capability, to describe the types of plant modification which have been identified as possible future requirements to be applied through rulemaking, which modifications can be readily made if required, and to show that the present design, together with procedures and training acceptable to NRC, meets all current NRC requirements such that the risk of adverse consequences of ATWS events is acceptably low and indeed minimal for full power operation of Summer pending the resolution of the possible ATWS modifications mentioned above.

Nuclear power plants have many diverse and redundant safety and control systems. These systems limit the consequences of abnormal operating conditions or an anticipated

transient. The vast majority of these abnormal operating conditions are of no safety significance because the various control systems function to return the unit to normal operation. For some of these abnormal operating conditions the various safety systems may be called upon to rapidly insert the control rods, or scram, thereby reducing the generation of heat. If there were a potentially severe "anticipated transient" and the reactor shutdown system did not "scram" as required, then an "anticipated transient without scram," or ATWS would have have occurred.

The concept of ATWS was first introduced in 1969 by a consultant to the Advisory Committee on Reactor Safeguards (ACRS). The NRC had available to them at that time the results of some very conservative calculations, based on conservative input and models, which showed that the consequences of an ATWS event could be severe for boiling water reactors in terms of peak reactor power and, for pressurized water reactors, peak reactor coolant system pressure. In September, 1973 the NRC issued WASH-1270, "Technical Report on Anticipated Transient Without Scram for Water Power Reactors." This report defined a probabilistic safety goal for ATWS events described on page 20. Having defined a safety goal, the NRC defined what they believed, at that time, to be a probability of an ATWS described on page 57. The staff also outlined the criteria that the various plants should meet and a requirement that new analyses be done as described on pages 68-90.

In September, 1974 Westinghouse submitted WCAP-8330, "Westinghouse Anticipated Transients Without Trip Analysis." This was the result of a significant model development effort and more refined definition of the appropriate input to be used in the analysis. This set of analyses showed that the consequences of an ATWS event were not as severe as previously believed. The peak primary coolant pressure was well below the NRC imposed limit of the ASME stress level C which corresponds to 3200 psi on the limiting components. Likewise, the potential for large offsite doses was very remote since the fuel was predicted to stay in nucleate boiling, i.e., no clad temperature excursion. All other criteria suggested by the NRC were likewise met for the Westinghouse design NSSS. These results were less limiting than those predicted in the NRC's report WASH-1270 due to more refined computer modeling. The NRC has since reviewed all of the modeling changes and has no outstanding questions concerning Westinghouse's calculation of the peak pressure or fuel failure limits during an ATWS event.

These results were dependent on the operation of only two systems for the limiting ATWS events, namely, the emergency feedwater system and the actuation of a main turbine trip. The operation or actuation of these systems is already provided by the plant's protection system. Because of the concern about common mode failures in the reactor protection SYSTEM, the NRC has proposed that these functions may need to be generated by new and diverse equipment. This

equipment would represent a minor modification to the Virgil C. Summer Nuclear Station.

Since the submittal of WCAP-8330, there have been other NRC, as well as Westinghouse, reports. These reports are listed in Attachment B. The Westinghouse reports describe the computer models used for ATWS analysis as well as additional sensitivity studies required by the NRC. The NRC reports describe their review of information given to them by Westinghouse as well as other light water reactor vendors.

Likewise, the Electric Power Research Institute (EPRI) undertook a major study concerning the probability of an ATWS event. These EPRI reports, referenced in Attachment B, demonstrate that the probability of an ATWS event is significantly lower than what the NRC had predicted (see reference 2, page 45). These results were based on plant operation data for the probability of an "anticipated transient" and a more detailed model to define the possibility of a common mode failure that would prevent scram.

The Westinghouse reports submitted during this time continued to conclude that the peak pressure would be well below the NRC suggested limit of ASME stress level C and the fuel rods would not experience a clad temperature excursion. Because this is not a temperature excursion, the fuel rods would not fail and would not lead to an offsite dose. The latest of these reports was submitted in December, 1979 (see reference 2). This latest report contains all of the modeling assumptions requested by the NRC. The peak pressure for the

reference limiting ATWS transient for Westinghouse systems on a generic basis was 2974 psia or 226 psi margin to the ASME stress level C limits. Similar calculations for three loop plants with model D steam generators, as in the Virgil C. Summer Nuclear Station, show a peak pressure of only 2785 psia or a 415 psi margin.

The Westinghouse conclusions have not changed significantly because of the inhere t design of the Westinghouse NSSS. This is best understood by a review of the limiting ATWS event which is a loss of external load from full power. This transient is assumed to be initiated by a turbine trip and a complete loss of main feedwater. This is generally referred to as a loss of heat sink transient which is characterized by the secondary side of the plant removing less energy than is produced in the primary loop. The reactor protection system quickly senses that this has occurred and rapidly scrams the control rods (negative reactivity) in to terminate the heat generation in the primary loop. The net result of transient is a minor pressure increase in the primary loop which is controlled by the various safety systems. For the ATWS case the transient is assumed to begin in the same manner but the NRC imposed assumption of a common mode failure somehow prevents the control rods from being inserted. Because of the loss of heat sink, in the ATWS case, the primary loop coolant temperature will begin to increase. This coolant temperature increase (less than 30°F) has the effect of inserting

negative reactivity, similar to scraming the control rods, but not as large of an effect, because the coolant temperature coefficient of reactivity is negative. The temperature coefficient is inherently negative because of the fuel design and becomes more negative as the fuel is burned up. The negative insertion of reactivity has the effect of reducing the core power level as would be done by scraming the control rod in, but in the ATWS case the power level is not reduced to zero. This reduced power level in the primary loop is now equal to heat removal by the secondary side and there is no longer a loss of heat sink. This would be the end of the transient except for the assumption that main feedwater is terminated at the beginning of the transient. Because of this, the initial inventory of water in the steam generators will continue to be boiled off until the "U" tubes carrying the primary coolant are uncovered. As this happens, the secondary side heat removal rate is once again reduced. This has the same effect as earlier in the transient, namely, increased primary loop temperature leading to a reduction in the power level. The difference at this time in the transient is that the rate of heat removal reduction is somewhat faster. This faster rate does not allow the system to remain in a thermo-dynamic equilibrium and finally causes the pressure in the primary loop to rise. The pressure rise is such that the pressurizer safety valves may not be able to limit the pressure to the safety valve setpoint. If this occurs, the pressure may rise to approximately 3000

psia (relative to a limit of 3200 psia) for less than one minute. This transient is mitigated by the actuation of the emergency feedwater system.

This transient is shown in graphical form, (1) the primary side power (core heat flux), (2) average coolant temperature, and (3) pressurizer pressure. These figures show rise in coolant temperature in two specific steps corresponding to the initial assumption of the turbine trip and the second is when the steam generator "U" tubes begin to uncover. The power level in the reactor core follows closely the coolant temperature, i.e., when the coolant temperature goes up, the power goes down. Finally, the resulting pressure in the primary loop is shown. These calculations were carried out to 600 seconds (10 minutes) at which time the plant operator is assumed to scram the control rods or start the safety injection system, either of which will bring the plant to a stable condition. These figures are from the Westinghouse report submitted to the NRC in December, 1979 which represents a typical plant which is more limiting than for the Virgil C. Summer Nuclear Station. This transient, which is the limiting ATWS event, was calculated assuming the control rods were never inserted into the core. During the course of this transient several different automatic trip signals would be generated that would scram the control rods and thereby prevent the pressure from going above its design valve of 2500 psia. Among these trips are:

Steam Generator Low-Low Reactor Trip

High Pressurizer Pressure Reactor Trip

High Pressurizer Level Reactor Trip

Overtemperature Delta-T Reactor Trip

The results of this transient are very sensitive to the magnitude of the moderator temperature coefficient of reactivity assumed. The value chosen for this particular analysis was consistent with the NRC guidelines (see reference 7, page E8) of a 95% value. This means that greater than 95% of the time the results of these ATWS calculations would result in lower peak pressures.

The NRC Staff's latest requirements are described in a report, SECY-80-409, "Proposed Rulemaking to Amendment 10 CFR Part 50 Concerning Anticipated Transients Without Scram (ATWS) Events," dated September 16, 1980. This report gives a history of this concern and the NRC proposal for resolution for all NSSS vendors. On page 5 of SECY-80-409, the NRC indicates that Westinghouse reactors have greater inherent ATWS MITIGATION capabilities than other LWR designs. The proposed resolution R48 is specific hardware modifications to be applied through rulemaking (see SECY-80-409 Appendix A).

The hardware modifications for Westinghouse designed NSSS's is ATWS Mitigation System Actuation Circuitry (AMSAC). This modification consists of additional logic (diverse from the existing Reactor Protection System) that would actuate the emergency feedwater system and trip the main turbine. These functions are similar to the functions that already exist within the Reactor Protection System.

The modifications required for AMSAC are not major and can be installed readily in a plant such as Virgil C. Summer Nuclear Station if they are finally required by the NRC.

The NRC plan to resolve the ATWS issue as discussed in their report SECY-80-409, includes proceeding with the rulemaking process for generic resolution. A schedule for this rulemaking has not been issued by the NRC.

In summary, the ATWS issue has evolved significantly since first being introduced to the ACRS in 1969. The bases for the early NRC report WASH-1270, i.e., that ATWS events have a significant probability and that the consequences are severe, have been addressed by not only Westinghouse reports, but as well as reports by EPRI. Hence, the conclusion of WASH-1270 no longer seems to be appropriate for a Westinghouse designed NSSS. Minor modifications, such as AMSAC, can reduce the already low risk of ATWS events for plants like Virgil C. Summer Nuclear Station.

As stated in the NRC Safety Evaluation Report, the Virgil C. Summer Nuclear Station will be subject to the Commission decision on ATWS. The basis for operation of the plant while final resolution is before the Commission is also discussed in the SER. In summary, the likelihood of severe consequences arising from the ATWS event is acceptably small and presently there is no undue risk to the public from ATWS. This is based on engineering judgement in view of (a) the estimated arrival rate of anticipated transients with potentially severe consequences in the event of scram

failure, (b) the favorable operating experience with current scram systems, and (c) the limited number of operating reactors. The NRC has concluded that pressurized water plants can continue to operate because the risk from ATWS events in the time period prior to final resolution is acceptably small (see Section 15.3.5).

To further reduce the risk from ATWS events, the NRC has required that emergency plant procedures be developed to train operators to recognize ATWS events and that the plant operators be trained to take APPROPRIATE MITIGATING actions ~~in the event of~~ IF an ATWS event OCCURS. Accordingly, I developed guidelines for the development of plant specific procedures for recognition of ATWS events and appropriate operator action. RWS

Plant emergency operating procedures for the Virgil C. Summer Nuclear Station have been reviewed by Westinghouse. These procedures address actions to be taken by the operators in the event an ATWS event has occurred. Specifically, manual reactor scram is initiated, emergency feedwater initiation is verified or initiated, if required, turbine trip is verified or initiated, if required, and emergency boration is initiated, if required.

It is my understanding that SCE&G is providing a separate affidavit which addresses procedures and training specific to the Virgil C. Summer Nuclear Station.

In summary, the risk of adverse consequences from ATWS events is acceptably low, indeed minimal, for full power operation of the Virgil C. Summer Nuclear Station and SCE&G

has met all current NRC criteria with regard to ATWS. In addition, the plant is amenable to possible ATWS modifications proposed by the NRC Staff for rulemaking.

I hereby certify that the foregoing information is true and correct to the best of my knowledge and belief.

Robert W. Steitler
ROBERT W. STEITLER

Subscribed and sworn to before
me this 6 day of May, 1981.

Rebecca L. Beynon (L.S.)
Notary Public

My Commission expires:

Rebecca L. Beynon, Notary Public
Monroeville Borough, Allegheny County
My Commission Expires Apr. 25, 1982
Member, Pennsylvania Association of Notaries

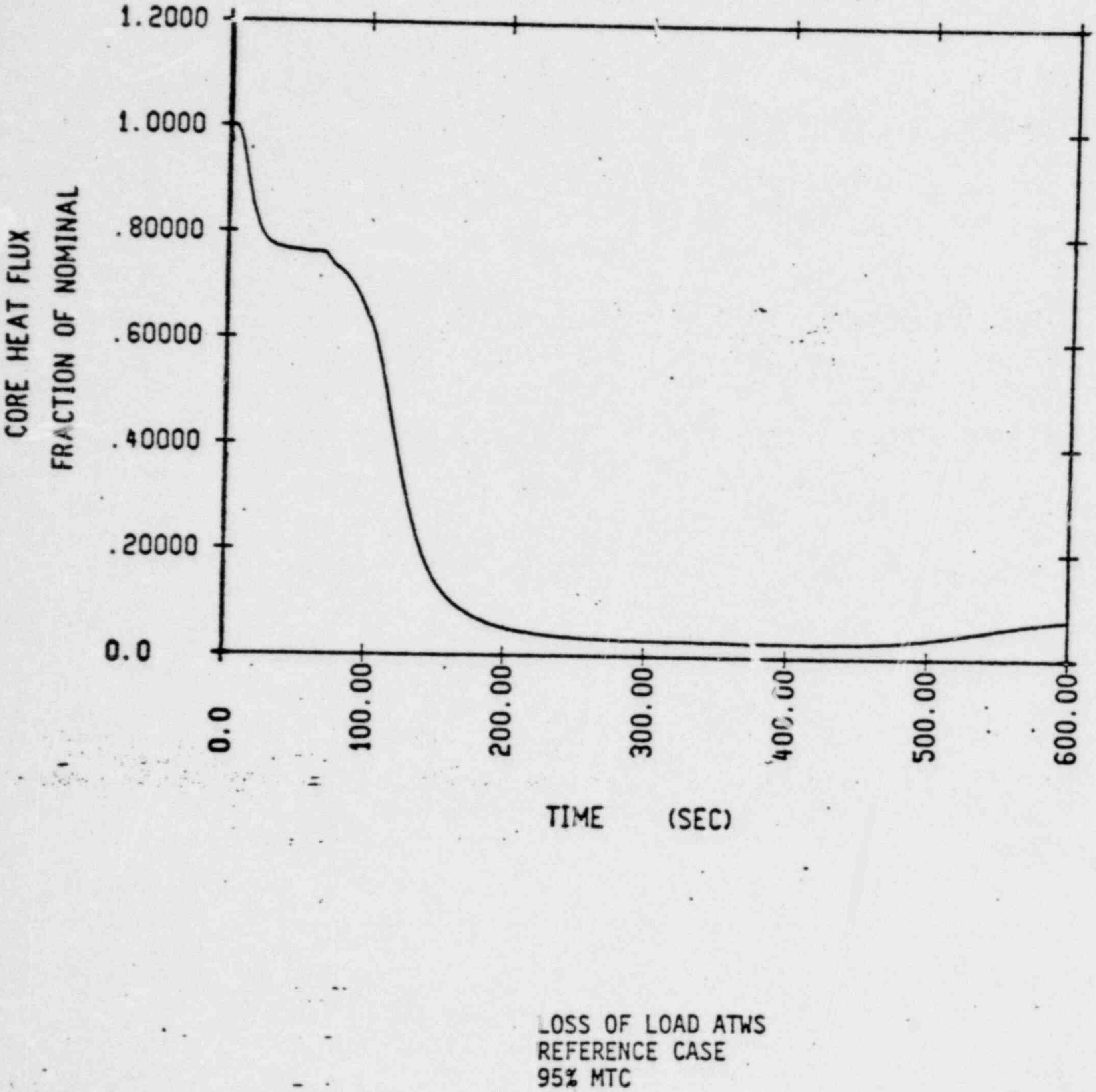
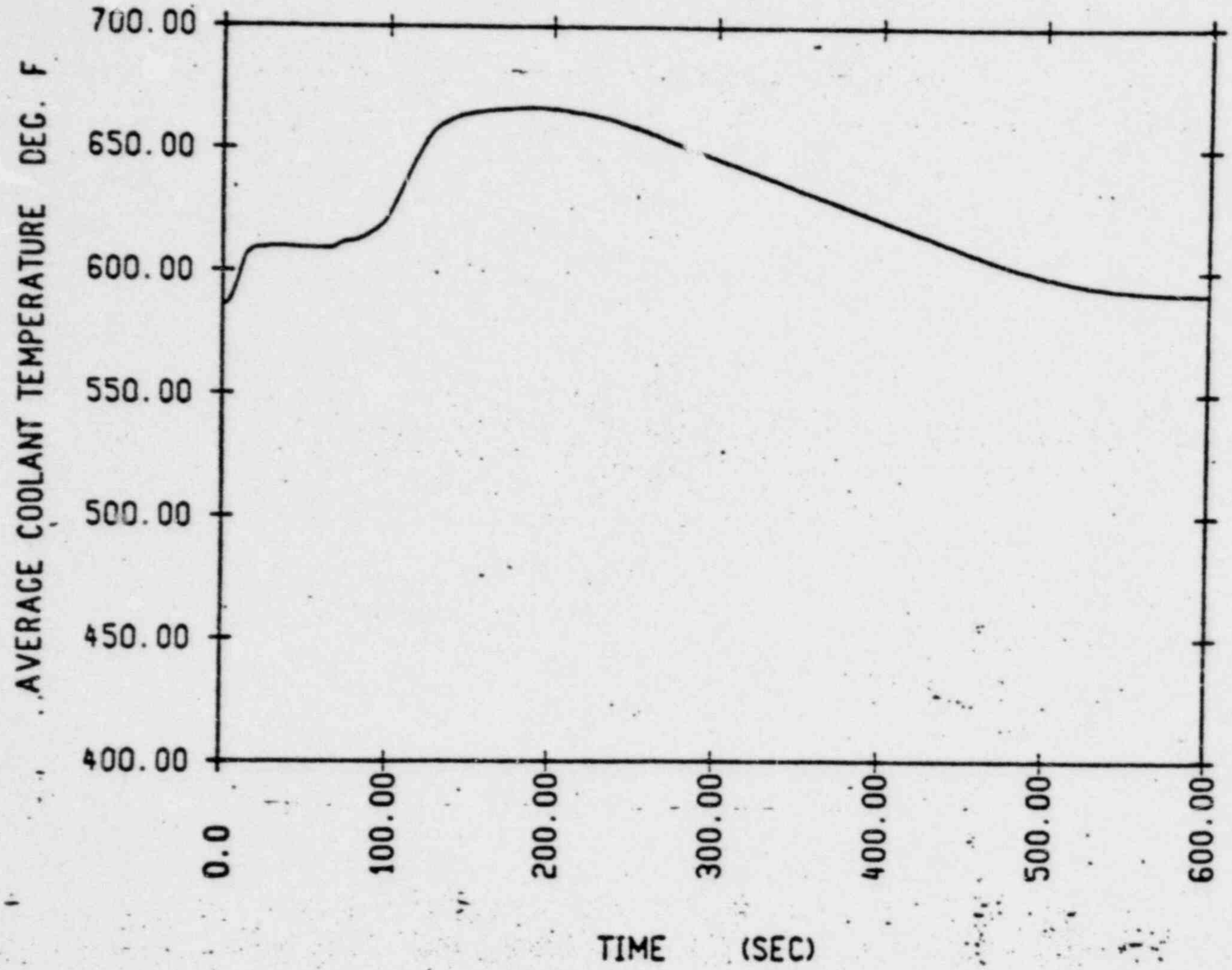


Figure 5.1-2

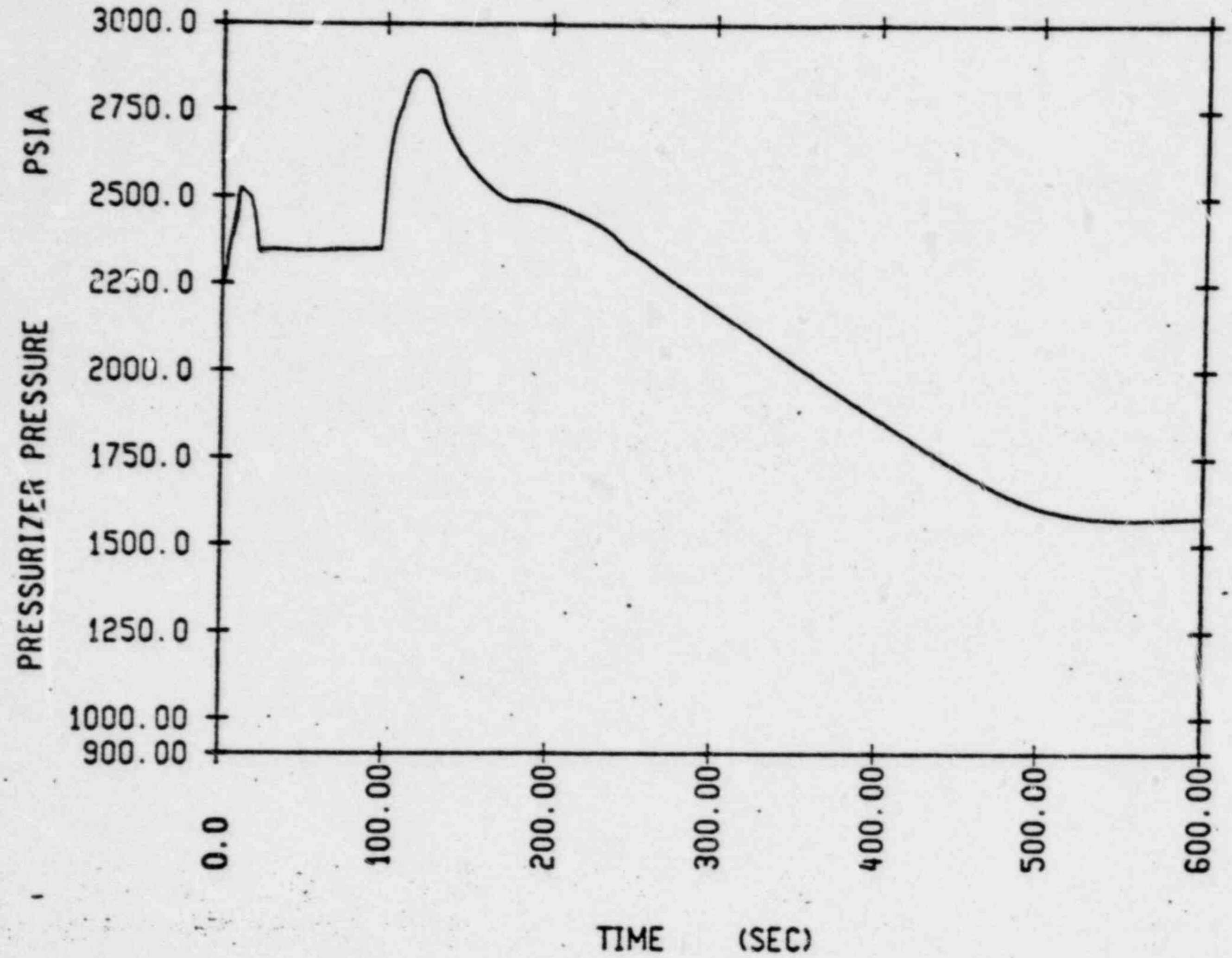
From NS-TMA-2182, December, 1979



LOSS OF LOAD ATWS
REFERENCE CASE
95% MTC

Figure 5.1-3

From NS-TMA-2182, December, 1979



LOSS OF LOAD ATWS
REFERENCE CASE
95% MTC

Figure 5.1-5

From NS-TMA-8182, December, 1979

ATTACHMENT A

PROFESSIONAL QUALIFICATIONS
ROBERT W. STEITLER
NUCLEAR TECHNOLOGY DIVISION
WESTINGHOUSE ELECTRIC CORPORATION

My name is Robert W. Steitler. My business address is Westinghouse Electric Corporation, Post Office Box 355, Pittsburgh, Pennsylvania 15230. I am Manager, Operating Plant Analysis, in the Nuclear Technology Division (NTD), Westinghouse Nuclear Energy Systems, Westinghouse Power Systems Company, and I have served in this capacity since September, 1977. My responsibilities include the management of activities that deal with transient analysis of non-LOCA events, setting design requirements for protection systems, developing models for ATWS analysis and transient analysis for an ATWS event for Westinghouse pressurized water reactor plants, and for reload safety analysis for operating plants.

I was graduated from Iowa State University with a Bachelor of Science degree in Engineering Science in 1968. From 1969 to 1971, I attended Carnegie Mellon University, where I received the degree of Master of Science in Nuclear Engineering.

I joined Westinghouse in 1968 as an Engineer in the Core Engineering group involved with core design and developing control system strategy.

I have been a member of the American Nuclear Society (ANS) and have helped develop the ANS standard for the calculational model to be used for ATWS events.

ATTACHMENT B

WESTINGHOUSE REPORTS:

1. Westinghouse Anticipated Transients Without Trip Analysis, WCAP 8330, August, 1974.

2. Letter from T. M. Anderson (Westinghouse) to S. H. Hanauer, with enclosures, December 30, 1979.

NRC REPORTS:

1. Anticipated Transients Without Scram for Light Water Reactors, USNRC Report, NUREG 0460, Volume 1, April, 1978.

2. Anticipated Transients Without Scram for Light Water Reactors, USNRC Report, NUREG 0460, Volume II, April, 1978.

3. Anticipated Transients Without Scram for Light Water Reactors, USNRC Report, NUREG 0460, Volume III, December, 1978.

4. Anticipated Transients Without Scram for Light Water Reactors, USNRC Report, NUREG 0460, Volume IV, March, 1980.

5. Letter from R. J. Mattson to All NSSS Vendors, February 15, 1979.

6. NRC Staff Report Presented to ACRS, December 9, 1975; "Status Reports On ATWS in Westinghouse Reactors".

7. SECY 80-409, Proposed Rulemaking to Amendment 10CFR Part 50 Concerning ATWS Events, September 16, 1980.

EPRI REPORTS:

1. ATWS: A Reappraisal, Part 1, An Examination and Analysis of WASH 1270. NP 251, August, 1976.

2. ATWS: A Reappraisal, Part 2, Evaluation of Societal Risk Due to Reactor Protection System Failure, Volume 3, PWR Risk Analysis, NP 265, August, 1976.

3. ATWS: A Reappraisal, Part 3, Frequency of Anticipated Transients, NP 801, July, 1978.