07/27/81

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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| In the Matter of | ) | |
| | ) | |
| HOUSTON LIGHTING AND POWER COMPANY | ) | Docket No. 50-466 |
| | ) | |
| (Allens Creek Nuclear Generating Station, Unit 1) | ) | |

NRC STAFF TESTIMONY OF MARVIN W. (WAYNE) HODGES
ON DOHERTY CONTENTIONS 8, 17 AND 32; TEXPIRG ADDITIONAL
CONTENTIONS 41, 53 AND 55; AND BOARD QUESTIONS 6 AND 17

Q. Please state your name and position with the NRC.

A. My name is Marvin W. (Wayne) Hodges. I am employed by the U.S. Nuclear Regulatory Commission as a Principal Reactor Engineer in the Reactor Systems Branch of the Division of Systems Integration. A copy of my professional qualifications is attached.

Q. What is the purpose of your testimony?

A. The purpose of this testimony is to respond to Doherty Contentions 8, 17 and 32; TEXPIRG Additional Contention 41, 53 and 55; and Board Questions 6 and 17. These contentions will be responded to separately below.

DOHERTY CONTENTION 8

> The petitioners are not adequately protected against Anticipated Transients Without Scram Accidents (ATWS). New information shows that 20 transients per annum are typical for new reactors with about 6 transients per annum typical after several years. Applicant has only a manually operated SCRAM system as its redundant system.

Q. What is ATWS?

A. Nuclear plants have safety and control systems to limit the consequences of temporary abnormal operating conditions or "anticipated transients." Some deviations from normal operating conditions may be minor; others, occurring less frequently, may impose significant demands on plant equipment. In some anticipated transients, rapidly shutting down the nuclear reaction (initiating a "scram") and thus rapidly reducing the generation of heat in the reactor core is an important safety measure. If there were a potentially severe "anticipated transient" and the reactor shutdown system did not "scram" as desired, then an "anticipated transient without scram," or ATWS, would have occurred.

Q. What is the status of the ATWS issue?

A. In December, 1978, Volume 3 of NUREG-0460, "Anticipated Transient Without Scram for Light Water Reactors" was issued describing the proposed type of plant modifications the Staff believed necessary to reduce the risk from ATWS events to an acceptable level. We issued requests for the industry to supply generic analyses to confirm the ATWS mitigation capability described in Volume 3 of NUREG-0460, and subsequently we presented our recommendations on plant modifications to the Commission in September, 1980. The Commission will determine the required modifications to resolve ATWS concerns as well as the required schedule for implementation of such modifications. ACNGS is subject to the Commission's decision on this matter.

Q. What is being done to provide significant protection against ATWS events in the interim?

A. It is our expectation that for operating plants the necessary plant modifications will be implemented in one to four years following a Commission decision on ATWS. As a prudent course, in order to further reduce the risk from ATWS events during the interim period before completing the plant modifications determined by the Commission to be necessary, we require that the following steps be taken:

(1) An emergency operating procedure should be developed for an ATWS event, including consideration of scram indicators, rod position indicators, average power range flux monitors, reactor vessel level and pressure indicators, relief valve and isolation valve indicators, and containment temperature, pressure and radiation indicators. The emergency operating procedures should be sufficiently simple and unambiguous to permit prompt operator recognition of an ATWS event.

(2) The emergency operating procedures should describe actions to be taken in the event of an ATWS including consideration of manually scramming the reactor by using the manual scram buttons, changing the operation mode switch to the shutdown position, stripping the feeder breakers on the reactor protection system power distribution buses, scramming individual control rods from the back of the control room panel, tripping breakers from plant auxiliary power source feeding the reactor protection system, and valving out and bleeding off instrument air to scram solenoid valves. These actions must be taken immediately after detection of an ATWS event. Actions should also include prompt initiation of the residual heat removal system in the suppression pool cooling mode to reduce the severity of the containment conditions; and actuation of the standby liquid control system if a scram cannot be made to occur.

(3) An automatic recirculation pump trip is to be included in the design to assist in minimizing pressure increase during a postulated ATWS through void reactivity feedback.

Early operator action as described above, in conjunction with the recirculation pump trip, would provide significant protection for some ATWS events, namely those which occur: (1) as a result of a common mode failure in the electrical portion of the scram system and some portions of the drive system, and (2) at low power levels where the existing standby liquid control system capability is sufficient to limit the pool temperature rise to an acceptable level.

Q. Does the Applicant have only a manually operated SCRAM system as its redundant system for the safe shutdown of the reactor as asserted in this Contention?

A. No. The reactor protection system (RPS) is a redundant, safety grade, multi-channel system. It is designed to meet current single failure requirements. It has a manual backup capability plus a separate manually actuated standby liquid control system (SBLCS) to inject neutron absorber solution into the core when other control systems are ineffective or inoperable.

Q. In Doherty Contention 7, it is also asserted that if low pressure coolant injection (LPCI) core spray water from the suppression pool is used following an ATWS event (assuming depletion of water from the condensate storage tank), it will increase core reactivity thereby causing high core temperature and increase the possibility of core melt. What is your response to this assertion?

A. The high pressure core spray (HPCS) initially takes suction from the condensate storage tank but the HPCS suction switches automatically to the suppression pool on either low condensate storage tank level or high suppression pool level. The low pressure core spray (LPCS) and low pressure coolant injection (LPCI) both take suction from the suppression pool.

During an ATWS event, the HPCS would inject water to make up inventory lost as a result of discharge through the safety/relief valves (SRVs). The discharge through the S' s would heat up the suppression pool. Thus, the suppression pool temperature would probably be higher than the condensate storage tank temperature at the time HPCS suction would be switched.

The LPCI and LPCS systems would not inject into the vessel during an ATWS unless the vessel was depressurized. The depressurization could occur as a result of actuation of the automatic depressurization system (ADS). Automatic actuation of the ADS requires concurrent signals of high containment pressure and low vessel water level. The high containment pressure could result from heat added from the SRV discharge and the low level could occur if HPCS and/or feedwater were not able to keep up with the inventory loss through the SRV discharge. The heat added to the suppression pool by the SRV discharge and the blowdown would likely cause the suppression pool temperature to be higher than the condensate storage tank temperature.

ATWS procedures require early actuation of the standby liquid control system to inject boron. Once the boron is injected, the core should be sufficiently subcritical that injection of water from either the suppression pool or the condensate storage tank should not cause a return to criticality. Prior to boron injection, the temperature coefficient of the moderator (including void coefficient) should be negative so that injection of suppression pool water should affect the criticality in the core less than injection of condensate storage tank water.

Q. What actions should be taken with regard to ACNGS at the CP stage?

A. Because the ATWS rulemaking is before the Commission and should be completed years before ACNGS would expect to operate, there is no need to take action at the CP stage. Once the ATWS rule is promulgated, ACNGS, as well as the rest of the LWR industry, will be required to comply with the rule. Therefore, nothing can be gained at this time either by speculating on the precise components which will be required by the rule or by debating the preferability of the alternatives which may be available. Should rulemaking not be completed by the time ACNGS would expect to operate, the interim approach described above could be used to provide significant protection against ATWS.

DOHERTY CONTENTION 17

> Intervenor contends pressure from blowdown following a Power Excursion Accident (PEA), Loss of Coolant Accident (LOCA) or Power Coolant Mismatch Accident (PCMA) combined with a single or several stuck relief valves may hit the suppression pool with sufficient force to permit escape of radioactive gases by causing cracks in the containment building wall and endanger Intervenor's health and generic safety interests. There has been considerable unreliability in pressure relief systems in BWRs, and the reduction from 22 to 19 relief valves increases the danger from failure of any single relief valve or more than one relief valve. Applicant should be required to research all data on such valves, and:

a) Commit to the use of one type with best record of performance during blowdown conditions; or

b) use a variety of manufacturers' products to prevent common mode failure.

Q. What are safety/relief valves and what is their function?

A. The safety/relief valves are multiple function valves discharging directly to the pressure suppression pool. The safety function includes protection against overpressure of the reactor vessel. The relief function is the same as the safety function except that the valves open at lower pressure upon command by a pressure signal. A subset of the safety/relief valves is used for rapid depressurization of the reactor vessel.

Q. Is the failure data on safety/relief valves for operating BWRs applicable to ACNGS?

A. Not directly. Most of the failure of safety/relief valves (SRVs) reported in NUREG-0462, "Technical Report on Operating Experience with BWR Pressure Relief Systems," are attributable to the design of pilot-operated SRVs. ACNGS will use direct acting Crosby valves which are not pilot-operated. The operating experience with Crosby valves is very limited, but the BWR Owners Group response to item II.K.3.16 of NUREG-0737, "Clarification of TMI Action Plan Requirements," indicates that the Crosby valve failure rate is expected to be at least a factor of 10 less than for the three-stage pilot-operated Target Rock valve, because of the elimination of the pilot.

Q. Does the reduction in the number of safety/relief valves from 22 to 19 for ACNGS increase the danger from failure of any single relief valve or more than one relief valve?

A. Failures of relief valves can be grouped as (1) failure to open, (2) spurious opening with failure to close, and (3) failure to close after opening on demand. Each group of failures will be discussed separately.

One or more relief valves may fail to open in the relief mode, but subsequently open in the safety mode. ASME code requirements allow credit for only half of the valves in the relief mode in overpressure analyses. Thus, up to half of the valves could fail to open in the relief mode and still maintain the vessel pressure at less than the code limit of 110% of design pressure for the most severe overpressure transient. If, in addition, one valve should fail in the safety mode, sensitivity studies on other BWRs show that the vessel pressure would only increase about 30 psi above the pressure for all safety valves opening.

The spurious opening of a relief valve during operation is an analyzed event which does not endanger either the fuel or pressure relieving system. The analyses are supported by numerous events on operating plants in which no damage was experienced. Spurious opening of relief valves has to date been more of an operational nuisance than a safety problem. Reducing the number of valves from 22 to 19 should reduce slightly the probability of a spurious opening.

Failure of a relief valve to close after opening on demand is similar to a spurious opening in terms of consequences. The peak blowdown load on the suppression pool occurs on the initial opening of the valves; subsequent loads on the discharge piping are less even if the valve sticks open. The high pressure core spray can maintain inventory above the top of the fuel and no fuel failure would result from the valve sticking open.

Q. Should valves from a variety of manufactures be used at ACNGS to prevent common mode failures?

A. Use of more than one make of valve would not preclude the possibility of common mode failure. Even with multiple valve design, there still exists a potential for common mode failures in the air supply to the valve actuators. Use of all valves of the same type simplifies maintenance and replacement of faulty valves, thereby reducing the chance of human error. Therefore, use of all valves of the same type is desirable.

DOHERTY CONTENTION 32

> Intervenor contends the vaporization rate of the Emergency Core Cooling System (ECCS) water during a design based loss of coolant accident (LOCA) is more rapid than the General Electric (GE) model predicted for an 8 X 8 fuel assembly core such as ACNGS. This was reported by the Advisory Committee on Reactor Safeguards from its March 8-10, 1979 evaluation meeting on the William Zimmer Nuclear Power Station, Unit 1 from data collected by GE using the two loop test apparatus. Further, Intervenor contends:
>
> a) any revision of the ECCS model should be used in determining the capacity requirements for the ECCS of ACNGS, and
>
> b) a calculation should be made to include the fact that the ACNGS channel boxes are of different thickness than the enclosures of the Zimmer plant, and
>
> c) the results of the revision and calculation should be applied in the construction of the ECCS of ACNGS.

Q. Why was there a concern that the General Electric (GE) ECCS model underpredicts the vaporization rate?

A. Comparison of blowdown tests run in the two loop test apparatus (TLTA) in 1978 raised Staff concerns about the conservatism of part of the ECCS evaluation model used by the GE Company. The TLTA configuration is a scaled BWR/6 design and includes the following major components: (1) pressure vessel and internals, (2) and 8 X 8 heated bundle, (3) two recirculation loops, (4) ECC systems (HPCS, LPCS, LPCI), (5) automatic depressurization system, and (6) auxiliary systems.

During August of 1978, test number 6405 was conducted. The test had an average power bundle with low ECC injection flow. Results of the test were compared with those from test 6007 which had the same initial conditions but no ECC injection. The comparison was presented in the monthly report issued in September, 1978 and in a program management group meeting on September 21 and 22, 1978. The comparison showed that the system depressurized more slowly with ECC injection than without ECC injection. Since the slower depressurization with ECC injection was contrary to intuitive expectations, GE was requested to discuss the test results and implications with the NRC.

Two theories were advanced as to why ECC injection slows the depressurization: (1) Additional steam is produced by ECC fluid contacting the core or hot vessel walls, and (2) increased liquid at the break decreases the volumetric break flow. The first theory led to the concern that the vaporization correlation used to predict steam updraft in the REFLOOD code might underpredict the actual steam updraft and result in a premature break-down of flooding due to counter-current flow at the top of the fuel assemblies. Also if the SAFE code underpredicted the vaporization in the vessel, the calculated depressurization rate would be too high and would result in early prediction of actuation of low pressure ECC systems. REFLOOD and SAFE are component parts of the GE ECCS evaluation model.

Q. Why do you now believe that there is no basis for concern that the GE ECCS model underpredicts the vaporization or steaming rate?

A. In a letter to W. D. Beckner (NRC) and Dr. M. Merilo (EPRI) from G. W. Burnette (GE), "Further Evaluation and Interpretation of BD/ECC-1A Data," July 31, 1979, GE presented analyses which show that there was increased liquid entrainment in the blowdown flow for the test with ECC injection. Also, the analyses showed that the steam flow through the steam separator above the core was lower with ECC injection than without injection (due to quenching of steam by the spray flow). Therefore, the analysis of the TLTA data shows that the difference in depressurization rate is due to the liquid entrainment in the break flow and not due to increased steaming in the core.

Two repeat tests were conducted in TLTA with (test number 6425) and without (test number 6426) ECC injection. For these repeat tests, improved break flow instrumentation was used to verify that the difference in depressurization rate was due to increased liquid in the break flow rather than increased core steam flow. As discussed in a letter from L. Harold Sullivan (NRC) to Paul S. Check, (NRC), "Status Request on Modeling Capabilities of the TLTA Experiment - 6506," February 23, 1981, the repeat tests clearly show that the liquid in the break flow is the reason for the difference in the depressurization rate. Therefore, the concern that the REFLOOD and SAFE codes are under-predicting the steaming rate is without basis, and the GE ECCS evaluation model continues to be acceptable.

Q. Do you see any need to revise the ECCS model and verify the capacity requirements for the ECCS of ACNGS?

A. The ECCS model used for ACNGS is adequate for verifying ECCS capability.

Q. Are the channel boxes for ACNGS of the same thickness as for Zimmer?

A. Both Zimmer and ACNGS have channel boxes which are 0.120 inch thick.

TEXPIRG AC 41

> Applicant's relief valve system against overpressurization is based on an analysis that is too close to ASME Boiler & Pressure Vessel Code allowable upper pressure limit for public safety. Although Applicant and Staff maintain recirculation pump trip will bring some pressure relief, the assumption of relief valve performance at ACNGS does not take poor performance of radiation monitors which signal high flux and actuate the pressure relief valves into account. *Nuclear Safety* 19(1), 1978, p. 82, shows 17 "Reportable Occurrences" for 1976 with such instruments among 22 BWR's, and *Nuclear Safety*, 20(1), 1979, p. 84, shows 36 such reports among 23 BWR's. The fact the high flux signal system is conservative to the high pressure signal is not significant if there is a high flux signal failure. Petitioner contends redundancy of signal systems is preferable.

Q. What is the role of the flux signal in an overpressurization transient?

A. The flux signal provides a reactor trip when the flux signal reaches a nominal 120% rated flux. For an overpressurization transient such as a Main Steamline Isolation Valve (MSIV) closure, the pressure wave resulting from the valve closure causes a short lived flux spike due to moderator void collapse. The flux spike is sufficient to trip the reactor.

Q. Are the safety/relief valves for ACNGS signaled to open on a flux signal?

A. No. The safety/relief valves are opened in the relief mode by a pressure signal and in the safety mode by system pressure exceeding a set spring pressure.

Q. Does the fact that there were 17 reportable occurrences for power range instruments in 1977 and 36 reportable occurrences for power range instruments in 1978 imply that those instruments are not accurate enough to provide dependable reactor trips?

A. No. Most reportable occurrences on instruments are for set point drift exceeding the allowable drift. The allowable drift for power range instruments is about 2% and measurement uncertainty is about 1%. The trip point assumed in transient analyses is increased over the set-points used in the plant to account for measurement uncertainty and drift. For an overpressurization event such as MSIV closure, the flux spike will peak at approximately 300% of nominal full power flux. The flux spike observed in the turbine t  p tests at Peach Bottom peaked at about 280%.

Q. Are there safety grade trips which are redundant to the flux trip?

A. Yes. A reactor trip on MSIV closure would occur prior to the flux trip. A high pressure reactor trip would occur slightly later than the flux trip and is available as a backup.

Q. Are there any other reactor trip signals?

A. Yes. A reactor trip would also be expected to occur from turbine stop valve closure; however, no credit is given for this valve position scram in analyses since its design does not meet the high quality standards provided by the safety grade reactor trip discussed above.

TEXPIRG AC 53

> This Intervenor contends Applicant should commit to a system to ascertain accurately how much non-condensible gas is in the reactor vessel, to assist in estimating the possible explosion hazard in the vessel during an ECCS. The need for this information was demonstrated at Three Mile Island, Unit 2, during its recent incident. Petitioner contends that inability to know accurately the amount of non-condensible gas in the reactor increases the chance of an explosion and damage to the fuel geometry and/or physical breaking of fuel rod clad.

Q. What non-condensibles could pose an explosion hazard in the reactor vessel?

A. Hydrogen and oxygen are the non-condensibles which have the potential for combining explosively inside the reactor vessel. The normal concentration of free hydrogen and oxygen is very small. Over heating of the fuel may cause oxidation of the cladding and generate free hydrogen.

Q. What conditions are required for generation of significant quantities of hydrogen?

A. Fuel rods must be uncovered for a long enough period of time, without core cooling, to oxidize a large fraction of the fuel cladding. This time may range from minutes for no cooling at all to hours with steam cooling available.

Q. This condition was experienced at TMI-2. Why shouldn't we expect a similar condition to exist at ACNGS sometime during the life of the plant?

A. This condition was experienced at TMI-2 because of confusion concerning the water level in the vessel which led to termination of high pressure injection. At TMI, water level was measured in the pressurizer, not in the vessel. ACNGS has unambiguous water level instrumentation for the vessel. Operator actions are keyed to maintaining or restoring water level in the vessel. Therefore, there is reasonable assurance that the operator will not terminate high pressure injection when fuel is uncovered.

Q. If large quantities of hydrogen were generated, would it remain in the vessel?

A. Most of it would be vented through the normal pressure control devices. If the break were small enough to cause the reactor pressure to remain high, non-condensibles, including hydrogen, would be vented through the safety/relief valves and RCIC turbine to the suppression pool. If the break were large enough to depressurize the reactor, then venting could occur through the break.

Q. Considering that the conditions required for generation of large quantities of hydrogen are not expected to exist and that there are vent paths for the hydrogen if it did exist, is there a need to measure hydrogen gas in the reactor vessel?

A. No.

<u>TEXPIRG AC 55</u>

> In the event of steam line break, rapid depressurization of the reactor vessel would take place resulting in frothing of the core steam bubbles and

> drawing of coolant water into the reactor. The movement of this water will cause an increase in reactivity before the SCRAM system will be effective. The reactivity insertion constitutes a danger to petitioner's health and safety because of the danger of fuel melt following such a power excursion. Petitioner contends Applicant must demonstrate the SCRAM system will function rapidly enough to prevent such increase in reactivity.

Q. Would the rapid depressurization which would follow a steam line break result in frothing of the core steam bubbles and drawing of coolant water into the reactor?

A. The rapid depressurization would cause some liquid in the vessel (e.g. core and lower plenum) to flash (i.e. some of the liquid would change to steam) thus causing a swelling of the two-phase level in the vessel. Some of the lower plenum water would swell into the core region. However, because the water in the core is saturated and the water in the lower plenum is subcooled, the water in the core would flash before the water in the lower plenum. Thus, there would be a delay before the lower plenum water would swell into the core region. This expansion of water in the lower plenum and other regions in the reactor vessel is known as "level swell."

Q. Would this "level swell" cause an increase in reactivity?

A. No. The level swell would result in a net increase of voids in the core and a decrease in reactivity and poses no danger of fuel melt.

BOARD QUESTION 6, RHR SYSTEM

> Staff and/or Applicant shall present evidence either to establish that the concerns of the ACRS are not applicable to ACNGS or that these concerns have been obviated by remedial measures. The ACRS concerns shall be treated as a Board Question.

Q. What are the concerns of the ACRS relative to the residual heat removal (RHR) system?

A. Two concerns have been expressed by the ACRS. First, in a letter to Dixy Lee Ray from W. R. Stratton, December 12, 1974, the ACRS expressed concern about the heat removal capability of the RHR system given a single failure in the let-down line for the shutdown cooling mode of operation. Second, in a letter from Dade Moeller to Marcus Rowden, December 1976, the ACRS expressed concern over the possibility of damage to a heat exchanger of the residual heat removal system by overpressurization or by hydrodynamic forces that could conceivably result from valve malfunction. This is associated with the steam condensing mode of operation of the RHR, or when the reactor core isolation cooling system is in use.

Q. What is the concern relative to the shutdown cooling mode of operation?

A. NRC implementation of General Design Criterion 34 requires that the capability exists to remove residual heat from the reactor core using only safety grade equipment and considering a single active failure. For the shutdown cooling mode of operation, the RHR system draws water from a single pipe which is connected to a recirculation loop. The ACRS concern is that failure to open a single valve in that pipe could incapacitate the shutdown cooling mode of operation.

Q. Given such a failure, can cold shutdown be achieved using only safety grade equipment?

A. Yes. The alternate shutdown cooling method is to flood the vessel to the elevation of the steamlines, open several of the automatic depressurization system (ADS) valves to discharge heated water to the suppression pool, and pump water back into the vessel from the suppression pool through the RHR heat exchanges. The RHR system operating in the low pressure injection mode would thus remove the residual heat discharged to the suppression pool via the ADS valves.

Q. Has the alternate shutdown cooling capability been tested?

A. The relief valves are being tested to demonstrate the capability to pass sufficient quantities of water to achieve cold shutdown. Calculations show that excess capability exists.

Q. What is the concern relative to the steam condensing mode of the RHR?

A. The RHR heat exchangers are of the shell and tube design. Primary water flows through the shell side (outside of tubes in a tube bundle) and cooling water flows inside the tubes. For the steam condensing mode of operation, water is drained from the shell side until the shell side level is about 75% of the level set point. Steam is admitted at the top of the shell and is condensed on the outer surfaces of the tubes in the tube bundle. If cold water were admitted to the shell side of the heat exchanger (as a result of a valve failure or operator error) during this mode of operation, the ACRS concern is that rapid condensation of the steam could result in water hammer in the heat exchanger. The water hammer could result in overpressurization of the heat exchanger and lead to unacceptable hydrodynamic loads on the heat exchanger and its supports.

Q. Has such an incident been reported?

A. Waterhammer has never been reported during the steam condensing mode.

Q. Has waterhammer occurred in any of the piping connected to the heat exchangers or in the heat exchangers themselves?

A. Yes. Valves in the steam lines connecting to the RHR heat exchangers sometimes leak. These leaks cause steam pockets to form in the top of the heat exchanger shell. Startup of a RHR pump then causes the steam to condense and a waterhammer results.

Q. Has this waterhammer caused any damage?

A. There have been instances of damage to snubbers, pipe hangers and welds which are attributed to waterhammer caused by condensing of the steam pockets.

Q. What is the NRC doing about this problem?

A. The ACRS's concerns regarding potential damage induced by hydrodynamic forces are being addressed as one of several types of water hammer effects being generically studied under Unresolved Safety Issue (USI) A-1, Waterhammer. Results to date indicate:

(1) Total avoidance of the potential for a waterhammer phenomenon is not practical.

(2) Observation of RHR system waterhammer damage has not appeared to result in unsafe conditions (i.e., loss of RHR function).

(3) Waterhammer during the steam condensing mode has not been observed.

BOARD QUESTION 17

> In light of a Board Notification of November 26, 1979 (BN-79-41), the staff shall present evidence as to the acceptability of using non-safety grade equipment for the mitigation of transients.

Q. What was the genesis of the concern discussed in BN-79-41?

A. In August 1979, Westinghouse informed their customers that the performance of non-safety grade equipment subjected to an adverse environment could impact the protective functions performed by non-safety grade equipment. These non-safety grade systems include:

(1) steam generator power operated relief valve control system,

(2) pressurizer power operated relief valve control system,

(3) main feedwater control system,

(4) automatic rod control system.

These systems could potentially malfunction due to a high energy line break inside or outside of containment. In IE Information Notice No. 79-22, the NRC also expressed concern that the adverse environment caused by the high energy line break could also give erroneous information to the plant operators.

Q. Did operating BWRs review their plants to determine if such a potential existed on their plants? If so, what was their finding?

A. Yes, the operating BWRs did review their plants for the potential and found no significant consequences. Currently, BWRs at the operating license stage are being asked to perform the same review.

Q. Why is this not a problem for BWRs?

A. Breaks outside containment, except for some instrument line breaks and a break in the scram discharge system, are isolated automatically. Thus, except for the effects of direct impingement, the environment resulting from the break should not be sufficiently severe to cause failure of control equipment. The instrument line break has been analyzed in Section 15.1.35 of GESSAR-238 PSAR and found to result in acceptable consequences. A break in the scram discharge volume has recently been considered for its effect on safety grade equipment. The probability of a break in the scram discharge volume has been calculated to be extremely low (on the order of $10^{-6}$ per reactor year, NUREG-0803, "Generic Evaluation Report Regarding Integrity of BWR Scram System Piping") and the operator has sufficient time and information to depressurize and thus reduce the effect of the break. Additionally, for Mark III containments, such as ACNGS, the scram discharge system is located inside the containment.

The principal non-safety grade equipment inside containment consists of the recirculation pumps, relief valve actuators and instrument lines. Failure of the recirculation pumps would have minimal impact on the transient resulting from a high energy line break inside containment; in fact, power is assumed to be lost to these pumps in analyses of breaks inside containment. NUREG-0515, SER § 6.3 (Supp. No. 2, March 1979) for Allens Creek. Spurious opening of one or more relief valves during a high energy line break would result in faster depressurization with less inventory loss; thus, the impact of such a failure would not be detrimental.

High ambient temperature resulting from a break inside containment could cause pressure signals to both safety and non-safety grade equipment to be in error. However, the time required to heat up the instrument lines is sufficiently long (approximately one-half hour) that no safety system actuations are affected and the operator is cautioned to look for this effect in his emergency procedures developed from the emergency procedures guidelines.

Marvin W. (Wayne) Hodges

Professional Qualifications

Reactor Systems Branch

Division of Systems Integration

U. S. Nuclear Regulatory Commission

I am employed as a Section Leader in Section B of the Reactor Systems Branch, Division of Systems Integration.

I graduated from Auburn University with a BS degree in Mechanical Engineering in 1965. I received a Master of Science degree in Mechanical Engineering from Auburn University in 1967.

In my present work assignment at the NRC, I serve as principal reviewer in the area of boiling water reactor systems. I also participate in the review of analytical models use in the licensing evaluations of boiling water reactors and I have the technical review responsibility for many of the modifications and analyses being implemented on boiling water reactors after the Three Mile Island, Unit-2 accident.

As a member of the Bulletin and Orders Task Force which was formed after the TMI-2 accident, I was responsible for the review of the capability of BWR systems to cope with loss of feedwater transients and small break loss-of-coolant accidents.

I have also served at the NRC as a reviewer in the Analysis Branch of the NRC in the area of thermal-hydraulic performance of the reactor core. I served as a consultant to the RES representative to the program management group for the BWR Blowdown/Emergency Core Cooling Program.

Prior to joining the NRC staff in March, 1974, I was employed by E. I. DuPont at the Savannah River Laboratory as a research engineer. At SRL, I conducted hydraulic and heat transfer testing to support operation of the reactors at the Savannah River Plant. I also performed safety limit calculations and participated in the development of analytical models for use in transient analyses at Savannah River. My tenure at SRL was from June 1967 to March 1974.

From September 1965 to June 1967, while in graduate school, I taught courses in thermodynamics, statics, mechanical engineering measurements, computer programming and assisted in a course in the history of engineering. During the summer of 1966, I worked at the Savannah River Laboratory performing hydraulic testing.