07/27/81

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

NRC STAFF TESTIMONY OF KAZIMIERAS M. CAMPE
RELATIVE TO THE SHELL PIPELINE COMPANY'S
6-INCH LIQUEFIED PETROLEUM GAS (LPG) PIPELINE

[Bishop Contention 6, Board Question 12]

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

A. My name is Kazimieras M. Campe. I am employed at the U.S. Nuclear Regulatory Commission as a Site Analyst in the Siting Analysis Branch. I have testified previously in this proceeding on aircraft and natural gas pipeline relocation hazards.

Q. What is the purpose of this testimony?

A. The purpose of this testimony is to respond to Bishop Contention 6 and Board Question 12.

Q. What does Bishop Contention 6 allege?

A. Bishop Contention 6 states as follows:

> The rupture of the six-inch liquid petroleum gas pipeline could cause a cloud of explosive gas to travel along depressions to the area of the plant before exploding with such force to damage the safety equipment at the plant and the workers at the plant. For this reason either the pipeline or the plant must be moved.

Q. Has the Staff addressed the hazards associated with the 6-inch LPG pipeline with respect to the proposed Allens Creek plant?

A. Yes. A discussion of the hazards associated with the p eline is provided in Supplement No. 2 (Section 2.2.4) to the Staff's SER for Allens Creek (NUREG-0515).

Q. What were the Staff's findings?

A. As noted in the Supplement, Staff perceived the possibility of extensive propane gas transport along Allens Creek due to the topographical depression formed by the creek. This is based on the fact that propane gas is 1.6 times denser than air, so that it would tend to flow downhill, along the creek from the location of the pipeline. The Staff indicated the need for assessing the extent of potential propane cloud formation and the fire or explosion hazard that may exist in the event of cloud ignition. The Staff also noted that some physical changes were feasible (e.g. pipeline relocation) if the liquified petroleum gas (LPG) hazard could not be shown to be inconsequential.

Q. Has the Staff performed any additional review of the 6-inch LPG pipeline since the issuance of Supplement No. 2 to the Allens Creek SER?

A. Yes. Since the issuance of Supplement No. 2, the Staff has requested and received from the Applicant revised and more detailed analyses of the release and transport of propane from a ruptured 6-inch LPG pipeline. The Staff also has been on a site visit for the specific purpose of examining the topographical features of the Allens Creek area.

Q. Has the Staff made any new findings as a result of the more recent review effort?

A. Yes. Following a more detailed review, the Staff has concluded that the potential effects of a fire or explosion hazards due to the release of propane gas from a ruptured 6-inch pipeline are estimated to be within acceptable limits with respect to safety related features of the proposed plant.

Q. What is the basis for the conclusion?

A. Our review focused on several distinct aspects of the Applicant's analyses of the postulated pipeline rupture. We reviewed the extent of a low lying propane cloud that potentially could be formed below the 140 foot isocline of Allens Creek. We also reviewed the potential for forming deflagrable and detonable propane clouds above the 140 foot isocline. Finally, we reviewed the potential thermal fluxes and overpressures stemming from propane deflagration and detonation, respectively. The Staff's findings with respect to the above reviews are that the estimated thermal fluxes and overpressures are within acceptable limits even though extremely conservative analyses were used.

Q. What are the specific considerations which were used to support the Staff's conclusions?

A. The first concern was to determine the extent of potential propane cloud formation along Allens Creek, since this was perceived as a means of bringing propane in significant quantities and concentrations close to the plant. At the Staff's request, the Applicant provided a propane vapor flow analysis which included gravity induced propane flow considerations that are based on some recently performed experiments with dense (negatively buoyant) gas plumes. The use of the experimental data was suggested since it supported the view that relatively little mixing can take place across the horizontal interface between a dense plume and the ambient and more buoyant fluid. This condition would tend to maximize the transport of propane along a channel such as Allens Creek, since losses to the upper layers of ambient air would be restricted. The use of this correlation led to the result that gravity flow of negatively buoyant propane could extend far enough along Allens Creek such that it would pass by the site. This estimate is in agreement with the initial perception that the Staff had with respect to the potential for forming extensive propane clouds as indicated by historical data of LPG pipeline failures.

Given the potential for propane flow along Allens Creek and past the site, the Staff reviewed the Applicant's analysis of propane transport from Allens Creek toward the plant. The Applicant assumed a 26,000 foot long line source to represent the propane within Allens Creek from the pipeline break to the cooling lake. Assuming 5 percentile meteorology and a wind blowing toward the plant, it was estimated that the maximum distances from the closest point on the line source (1800 feet between the plant and the closest point on Allens Creek) to the lower deflagrable and detonable concentrations were 220 feet and 190 feet, respectively. In other words, the deflagrable propane mixture would approach to within 1580 feet from the plant and the detonable mixture would come to within 1610 feet from the plant. The Staff has reviewed the Applicant's calculations and independently verified the parameters used in the calculations.

Finally, the potential fire and explosion effects on the plant in the event of cloud ignition were estimated. Based on the maximum extent of detonable concentrations of propane toward the plant and a line source length of 26,000 feet, the estimated propane inventory available for detonation is 4.3×10^6 cubic feet. This is equivalent to 5.4×10^4 pounds of TNT. Assuming that the entire inventory of propane is detonated at the closest point to the plant, that is at 1610 feet from the plant, the resulting peak reflected overpressure is about 2.2 psi.[1/] This is within the design basis overpressure criteria for the plant safety related structures. Accordingly, it is the Staff's conclusion that the plant can safely withstand the potential detonation effects due to propane clouds associated with the 6-inch LPG pipeline.

With respect to fire effects, the estimated propane inventory within deflagrable limits is 6.1×10^6 cubic feet. Even if the entire propane inventory available for deflagration were to be located near the closest point to the plant, that is about 1580 feet from the plant, the maximum thermal flux at the plant would not exceed 31 kW/m^2. It would take about six hours of exposure at this rate before any significant effects were produced on plant structures. This supports the Staff's conclusion that the plant is safe from any thermal effects produced by the postulated propane clouds.

1/ The Staff's estimate of the overpressure due to propane cloud detonation is higher than the Applicant's estimate for the following reasons: (1) The Applicant used an overpressure versus scaled distance correlation reported by Brode. This correlation yields overpressures which are about a factor of 2 lower than those determined by the Staff. The Staff's estimates are based on Kingery's data which are derived from nuclear explosions. (2) The Applicant used the peak incident overpressure in assessing the effects of propane detonation. The Staff believes that the use of the peak reflected overpressure is more appropriate. The reflected overpressure is a factor of 2 higher than the incident overpressure.

Q. In reference to Board Question 12, has the Staff assessed the hazards associated with gases other than LPG?

A. The Staff has not determined what other hydrocarbons may be carried within the 6 inch pipeline. However, the assumption that it carries LPG is reasonable since other hydrocarbons that potentially could be carried within the pipe will not pose a greater hazard.

Q. What is the basis for this conclusion?

A. The magnitude of a hydrocarbon vapor cloud hazard is determined by two principal factors. First, the density of the gas relative to air determines the extent of the cloud. The density of propane was found to be sufficiently high such that gravity induced flow could be postulated to fill Allens Creek down to the cooling lake. Hydrocarbons with higher densities would of course have the same potential for filling the creek.

Secondly, the TNT mass equivalency varies from one hydrocarbon to the next. However, we have assumed the highest value of 240% which actually corresponds to methane. For propane the value is 228% which is typical of most other flammables such as isobutane, butadiene, and propylene.

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SUPPLEMENTAL TESTIMONY OF
TAI L. HUANG REGARDING REACTOR WATER LEVEL INDICATORS

[Doherty Contention 41 and
TEXPIRG Additional Contention 54]

Q. Please state your name and position with the Nuclear Regulatory Commission.

A. My name is Tai L. Huang. I am employed as a Nuclear Engineer in the Thermal-Hydraulic Section of the Core Performance Branch, Division of Systems Integration, Office of Nuclear Reactor Regulation. A statement of my professional qualifications is attached (Attachment A) to this testimony.

Q. What is the purpose of your testimony?

A. The purpose of my testimony is to respond to two consolidated contentions which are basically concerned with the possibilitiy of spurious water level indication at Allens Creek based primarily on incidents at Three Mile Island and Oyster Creek. The consolidated contention (Doherty 41 and TEXPIRG Additional Contention 45) reads as follows:

> Intervenor's health and safety interests are endangered due to inadequate water level indicators for the reactor vessel for the proposed atomic

plant. That such indicators are often defective and mislead operators into actions which aggravate reactor incidents are evidenced by two recent incidents at U.S. facilities. At Three Mile Island, Unit II, spurious water level indications in the pressurizer and the reactor vessel resulted in operator errors which aggravated the event (March 29, 1979); and spurious water level indications in the Oyster Creek Nuclear Power Plant, resulted in operators failing to take action until the water level was dangerously low (May 2, 1979) - specifically the operator failed to open valves which would have allowed coolant to be pumped from the condensor to the reactor vessel. Intervenor contends Applicant must develop an alternative whereby the water level is sensed more reliably by redundant as to type level indicators and redundant as to function water level indicators. Intervenor contends an accident where a core uncovering results from unreliable water level sensing can lead to a release of radioactivity in excess of 10 C.F.R. 100, endangering his health and safety interests.

Intervenor further contends that inadequate water level indicators will lead to serious accidents for ACNGS, as at Three Mile Island, because the reactor systems are sufficiently similar in design being both dependent on safety systems actuated when reactor water level threatens to reach the top of the fuel rods. Because the proposed ACNGS has a higher power core density than any BWR this contention is particularly relevant to this proceeding. The Oyster Creek event provides a basis for showing much of the accident sequence has occurred in a BWR system.

Q. Is the reactor water level indication system to be installed at Allens Creek the same or similar to that employed at Three Mile Island, Unit 2?

A. No. The systems are completely different.

Q. Would you please explain the typical water level indication system which has been used in PWRs?

A. For TMI and other PWRs, the normal water level range in the reactor coolant system is within the pressurizer and is maintained by the pressurizer control system. Under normal circumstances, if there is some level indication in the pressurizer, the rest of the system should be full of coolant. However, under TMI conditions, i.e., stuck open PORV, steam flow into the pressurizer prevented drainage of pressurizer coolant such that the pressurizer indicated a water level while the primary coolant system was not full.

Q. What is different about the system you have just described compared to the one which will be installed at Allens Creek?

A. It should be apparent from the previous answer that PWRs presently have no reactor water level instruments in the reactor vessel itself. However, all BWRs, including Allens Creek, have pressure taps placed inside the reactor vessel so that vessel level indications can be received by the operators in the control room.

Q. Explain briefly how the BWR water level indication system operates.

A. In BWRs, water level is measured by the operation of differential pressure sensing devices which have had a long and reliable inservice history in BWRs. Condensing chambers connected to the steam space in the reactor vessel are used as the reference leg. Pressure taps at different levels in the water space of the reactor vessel are used as the variable leg sensing taps for narrow and wide range instruments. Narrow range instruments and associated control room indicators and recorders monitor water level approximately between the bottom of the steam dryer and bottom of the steam separator. Wide range instruments and associated control room indicators and recorders monitor water level approximately between the bottom of the steam dryer and the top of the core.

The differential pressure in the two legs permits determination of reactor pressure vessel water level, since the water level is a function of the differential pressure.

Q. Are the pressure sensing devices and associated control room indicators and recorders described in your previous answer fully redundant as to function.

A. Yes. There are eleven separate differential pressure sensing channels and control room indicators and recorders. Each water level range in the reactor vessel is overlapped by more than one separate sensing/indicating channel. There are two wide range level indicators/recorders and one wide range indicator, one narrow range level indicator/recorder and three narrow range indicators, one fuel zone indicator and an indicator/recorder, a high water level upset range indicator/recorder (overlaps the narrow range and wide range indicators and recorders) and a shutdown wide range level indicator (overlaps the upset range recorder).

The narrow range instruments are used to indicate water level for normal plant operation and the wide range instruments are used for ECCS initiation as a result of a low water LOCA transient. All of differential pressure devices and associated readout instruments in the control room will have to comply with the applicable provisions of Regulatory Guide 1.97, Revision 2, specifically those set forth in Part C, Section 1.3.1, "Design and Qualification Criteria-Category 1." These criteria include, among others, redundancy, single failure protection, and environmental and seismic qualification.

Q. If BWRs all have in-reactor vessel pressure taps for direct water level indication, and Oyster Creek is a BWR, why couldn't the spurious water level indication incident which occurred at Oyster Creek occur at Allens Creek?

A. Oyster Creek is a BWR-2 plant. The reactor coolant flow path in that design is through the annulus, recirculation lines and core area. For the level instrumentation to work properly, there must be an unrestricted and direct flow path between the annulus and core area so that the level indication will be consistent in both areas. For a non-jet pump reactor design such as Oyster Creek, there is a circumferential core shroud which acts as a buffer and restricts good fluid communication between the annulus and core region when the recirculation pumps are not running. (See Attachment B). Since the pump is not running, in a reactor scram the water level in the annulus might be higher than in the core because the steam generated by decay heat would condense back into the annulus region (but not into the core). The water level indication system at Oyster Creek has since been modified to eliminate this problem. However, for a jet-pump BWR-6 reactor design, such as Allens Creek, there is always good fluid communication between the two regions, since nothing restricts water flow whether the recirculation pumps are operating or not. (See Attachment C). Therefore, the reactor level instruments for Oyster Creek could provide a discrepant vessel level indication, while for Allens Creek there will always be a consistent and accurate level indication for both regions.

Q. What are your conclusions regarding this contention?

A. The reactor water level indication system to be installed at Allens Creek is different in critical respects from those used at TMI-2 and Oyster Creek, and the incidents at those facilities provide no cause for concern over the adequacy of the Allens Creek design. The Staff is confident that the water level indication system at Allens Creek will perform its intended function properly because:

(1) it is based on pressure taps in the reactor itself and differential pressure sensing devices which have been used reliably in BWRs for many years

(2) it is employed in a reactor design which eliminates the possibility of discrepant level indication in the annulus and core areas

(3) it will be designed in accordance with the stringent provisions of Regulatory Guide 1.97, Rev. 2.

Attachment A

Professional Qualifications

Tai L. Huang

July 1981

I am presently employed with the U.S. Nuclear Regulatory Commission as a Nuclear Engineer in the Thermal-Hydraulics Section of the Core Performance Branch, Division of Systems Integration, Office of Nuclear Reactor Regulation.

In my present work assignment at the NRC, I am responsible for the review of the reactor core thermal-hydraulic design, reload, and the functional requirements for core monitoring systems to provide capability for detection and response to inadequate core cooling conditions. I was responsible for the review of the thermal hydraulic aspect of containment designs in my previous work assignment at the NRC.

Prior to joining the NRC Staff in March, 1975, I was employed by Boeing Company as a Senior Mechanical Engineer (from 1972 to 1975). I was responsible for the thermal and fluid flow analysis for improving the aircraft engine performance and the environmentally controlled system design.

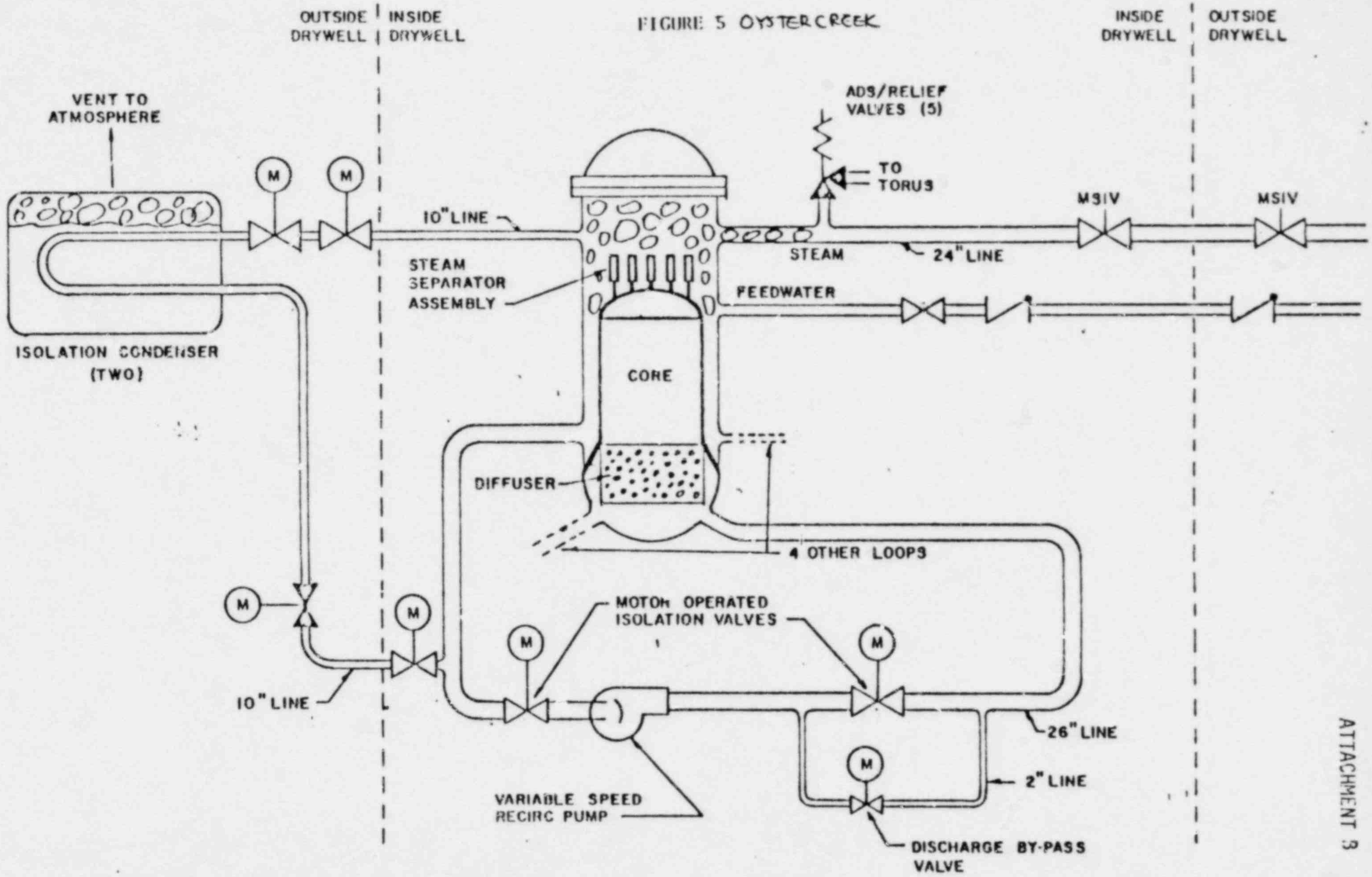
In 1972 I was employed by the Radiation Biology Laboratory of Smithsonian Institute as a Mechanical Engineer to be in charge of environmentally controlled chamber design.

In 1971 I was employed by the Research Laboratory for Engineering and Science (RLES) of the University of Virginia as a Senior Scientist to investigate the thermal-hydraulic properties of fluids.

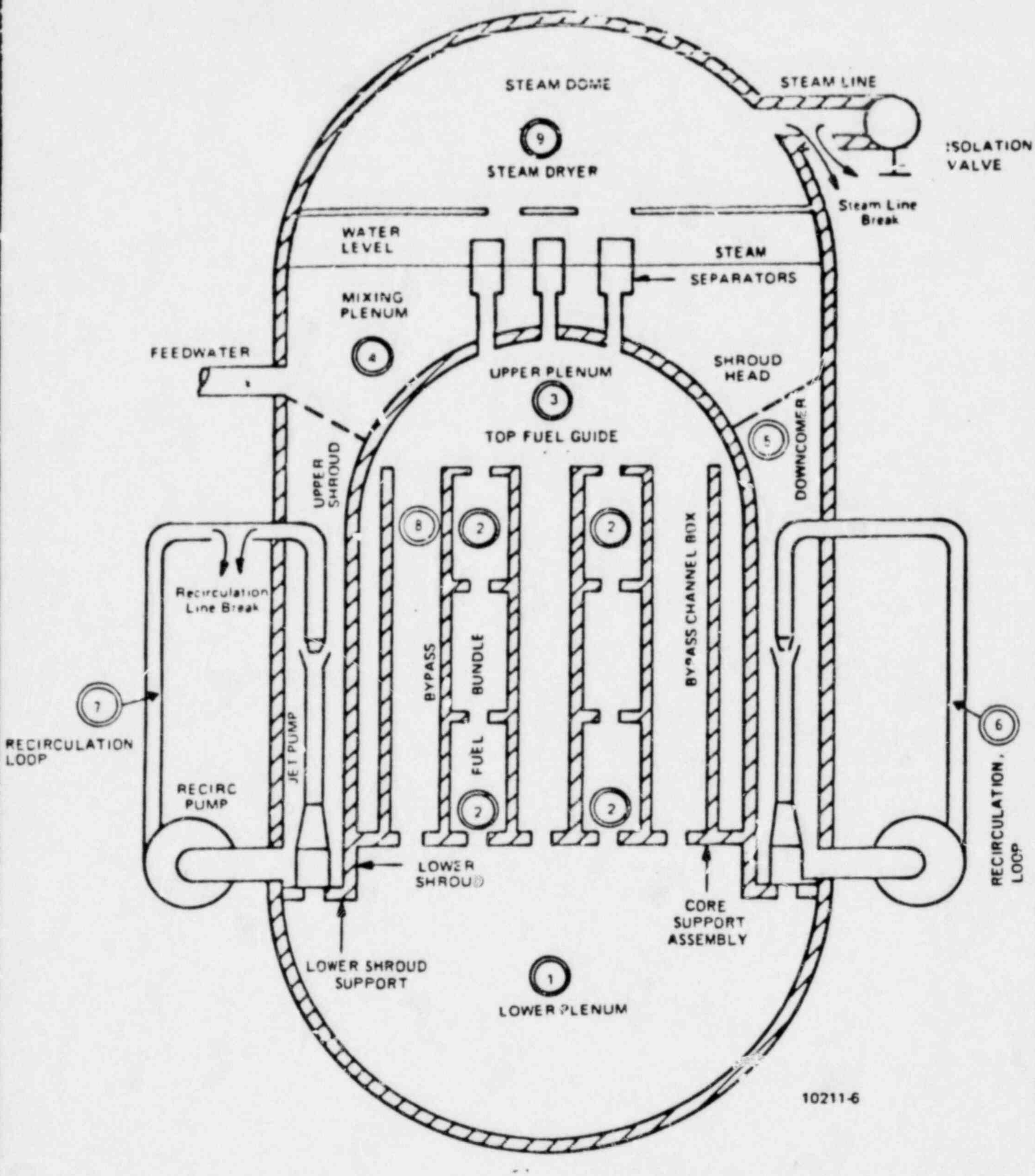
I graduated from the University of Virginia with a Ph.D. degree in Aerospace Engineering, 1970. I received a M.S. degree in Mechanical Engineering from the University of Iowa, 1967 and a B.S. degree in Mechanical Engineering from Cheng Kung University, Taiwan, 1964. I am a registered Professional Engineer in the State of Maryland.

ATTACHMENT B

FIGURE 5 OYSTERCREEK



ATTACHMENT C



HOUSTON LIGHTING & POWER COMPANY
Allens Creek Nuclear Generating Station
Units 1 & 2

PRESSURE NODES USED FOR DEPRESSURIZATION ANALYSIS

FIGURE 4.2-8