January __, 1985

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
THE CLEVELAND ELECTRIC	)	Docket Nos. 50-440
ILLUMINATING COMPANY, ET AL.	)	50-441
	)	
(Perry Nuclear Power Plant,	)	
Units 1 and 2)	)	

AFFIDAVIT OF CHARLES B. JOHNSON

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AFFIDAVIT OF CHARLES B. JOHNSON

Charles B. Johnson, being duly sworn, deposes and says as follows:

1. My name is Charles B. Johnson. I am employed by S. Levy, Inc., as a General Manager and Engineering Consultant. My business address is 3425 So. Bascom Ave., Campbell, CA 95008. I have been actively involved in the study and design of core cooling systems in Boiling Water Reactors ("BWRs") for approximately twenty years. I have also studied the detection and control of loss of coolant accidents in BWRs extensively. For example, I have led and participated in the preparation of two major reports for the Boiling Water Reactor Owners Group ("BWROG"): Review Of BWR Reactor Vessel Water Level Measurement Systems, July 1982 (SLI-8211), and Inadequate Core Cooling

Detection In Boiling Water Reactors, November 1982 (SLI-8218). A more complete statement of my professional qualifications is attached hereto as Exhibit A. I have personal knowledge of the matters set forth herein and believe them to be true and correct.

INTRODUCTION

2. The purpose of this affidavit is to address the contention of Ohio Citizens for Responsible Energy ("OCRE") that in-core thermocouples should be installed at Perry Nuclear Power Plant ("PNPP") as an additional method of detecting inadequate core cooling ("ICC"). As admitted in this contention, Issue 14 states:

> Applicant has not demonstrated that the Perry Nuclear Power Plant will meet regulatory safety requirements unless it installs in-core thermocouples, as suggested by staff regulatory guidelines including Regulatory Guide 1.97, Rev. 2.

The Atomic Safety and Licensing Board's Memorandum and Order of October 29, 1982 indicates that OCRE's "bases for the contention are the Reg. Guide [and NUREG 0737], plus an analysis performed by Battelle Laboratories and described in a letter by C. L. Wheeler and The Accident Hazards of Nuclear Power Plants, by Dr. Richard E. Webb, at 59-61." This affidavit will demonstrate that OCRE's contention is not well founded for three reasons. First, existing instrumentation at PNPP is

sufficient to adequately detect ICC. Second, in-core thermocouples suffer serious limitations and can actually contribute to the problems they are supposed to detect. Finally, the bases for OCRE's contention are inapplicable to the detection of ICC at PNPP.

3. My statements in this affidavit are based in part on the findings, conclusions and recommendations of three major reports prepared by S. Levy, Inc. for BWROG, of which the Cleveland Electric Illuminating Company ("CEI") is a member. Two of these reports are referred to in paragraph 1; the third is Thermal Analysis of In-Core Thermocouples in Boiling Water Reactors, December, 1981 (SLI-8121). SLI-8121 was prepared for BWROG in November of 1981 in response to the NRC Staff's position at that time that in-core thermocouples were necessary to detect ICC in BWRs. SLI-8121's findings confirmed BWROG's reservations regarding in-core thermocouples, which were that in-core thermocouples suffer serious time delays in detecting ICC, that inordinately large numbers of them would be required to provide reliable information about core temperature, and that they would provide ambiguous or inaccurate readings in certain situations.

4. SLI-8121's findings were presented to the NRC Staff. It was then agreed by the Staff and BWROG that a general study of ICC detection in BWRs would be appropriate. The result was SLI-8218, which was a detailed study of the detection of trends

toward ICC, the existence of ICC, and the return to adequate core cooling, in order to determine whether additional instrumentation for detecting ICC was warranted. The third report identified above, SLI-8211, was prepared for BWROG in response to a commitment which it made to the NRC Staff to fully evaluate the need for and desirability of design changes in existing BWR water level measurement systems. That commitment arose from a consensus between members of BWROG and the NRC Staff that, although the water level measurement systems used in BWRs such as PNPP have performed without any serious difficulties, a critical review might reveal some areas for potential improvement. The analysis, conclusions and recommendations of these three reports are discussed in some detail below.

5. In sum, the three reports referred to above, as well as analyses of the existing instrumentation at PNPP for monitoring ICC and of the stated bases for OCRE's contention, show that the installation of in-core thermocouples to detect ICC at PNPP is not warranted. ICC can be adequately detected by PNPP's existing, redundant reactor coolant level monitoring system and radiation monitoring systems. In-core thermocouples would not detect ICC as quickly as existing PNPP instrumentation, and are, unlike existing PNPP instrumentation, incapable of detecting decreases in coolant level indicating that ICC is about to occur. If in-core thermocouples were installed in numbers sufficient to give an accurate, complete indication of

localized overheating in the core, they would contribute to localized overheating by interfering with heat transfer or coolant flow, depending on their location. In-core thermocouples provide ambiguous data under certain circumstances, and do not significantly reduce the likelihood, which is already small, that existing PNPP instrumentation will not detect and trigger response to ICC. Moreover, since the issuance of Reg. Guide 1.97, Rev. 2, and the TMI Action Plan (NUREG-0737), the NRC Staff has modified its position and no longer calls for the use of in-core thermocouples in BWRs. Finally, the analysis by Dr. Webb cited by OCRE as a basis for the contention is based on assumptions which are inaccurate or inappropriate to the detection of ICC at PNPP.

PNPP'S ABILITY TO DETECT ICC

Definition of ICC

6. ICC is a condition in which the heat produced by the fuel rods is not removed by the reactor coolant rapidly enough to avoid overheating and damage of the fuel and the fuel cladding. At fuel temperatures of 1300°F to 1500°F, the consequence of overheated cladding is primarily the release of gaseous fission products into the reactor coolant. A maximum cladding temperature for the average fuel bundle of 1300°F is considered to be the onset of ICC. ICC can occur as a result of a loss of coolant accident ("LOCA") in which enough reactor coolant is lost so that adequate core cooling can no longer

take place. This situation would cause generalized ICC throughout the core. Localized overheating can also take place, due to blockage of reactor coolant flow, even when the reactor core as a whole is adequately cooled. While localized overheating is a form of ICC, it will be considered separately from generalized ICC in this affidavit. Therefore, references to ICC in the remainder of this affidavit will mean generalized ICC, unless localized overheating is specifically discussed.

Description of Existing Plant Instrumentation

7. The principal method by which PNPP monitors ICC is through measurement of the water level inside the reactor. This is a simple and reliable method of monitoring ICC because ICC cannot occur when the fuel rods are covered with water. During normal operation the water level in the reactor extends from the bottom of the reactor (below the bottom of active fuel) to approximately 18 feet above the top of active fuel. This provides approximately 18 feet of "buffer", i.e., cooling water which could be lost before ICC could occur. However, as discussed below, PNPP instrumentation can detect drops in the water level well in advance of the time the water level reaches the top of the active fuel.

8. PNPP uses differential pressure instruments to measure water level in the reactor core. Each set of instruments includes a "reference leg" which is a tube of water that

maintains a constant level above the reference point, and a "variable leg" which is connected to the water in the reactor. The level of the water in the variable leg is the height of the water in the reactor above the reference point. Both legs pass from the reactor vessel through the drywell and the drywell wall where they are connected to a differential pressure sensor. See Figure 1. The differential pressure sensor between the two legs compares the pressure exerted by the water in the variable leg to that exerted by the reference leg. Because the height of the water in the reference leg is fixed, the variable leg pressure can be converted directly into a measurement of height, which is done automatically by PNPP's instrumentation. What this actually measures is the "collapsed" water level inside the reactor, i.e., the level the water would have if all the steam in the reactor was above the liquid water. In reality, the cooling water contains some steam so the liquid surface is somewhat higher than it otherwise would be. Since the height of the reactor coolant (including any steam in it) is what is important in determining whether the reactor is adequately cooled, measuring collapsed water level is a conservative method of monitoring reactor water level and ICC.

9. PNPP uses eleven of these variable legs, each of which is connected through one or more differential pressure sensors to a reference leg, to measure water level. Four of

the eleven variable legs measure the water level over a "narrow range" above the top of active fuel, which brackets the normal operating water level (from approximately 19 to 13 and one-half feet above the top of active fuel). Another four measure the water level over a "wide range" (above the top of active fuel) from approximately 19 feet above the top of active fuel, to the top of active fuel. The range which each of these eight variable legs measures overlaps with the range measured by other variable legs, so that multiple measurements of all water levels are provided. The differential pressure sensors are connected to analog trip units. As the water level drops, the analog trip units activate various plant systems, such as high pressure core spray, feed pumps, recirculation pumps, isolation valves and so forth. Water level below the top of the active fuel is monitored by the remaining three separate and independent "fuel zone" variable legs which measure water level between the top and bottom of active fuel. See Figure 2. Differential pressure sensors in the wide range, narrow range, and fuel zone are also connected to gauges or recorders in the control room, providing the operator with visual indications of water level.

10. In addition to the differential pressure instruments used to detect ICC, PNPP has instrumentation which can be used to detect localized overheating. Localized overheating can occur when the core is covered with water, if for some reason

the flow of coolant through a particular fuel bundle is blocked. If this occurs, however, the overheated fuel rod releases gaseous fission products which can be detected with redundant steamline radiation monitors (which would automatically cause a scram, reducing power to decay levels). This system is sensitive enough to detect the release of fission products from any one of the 748 fuel bundles at PNPP. The steamline radiation monitors would detect the release of fission products and would initiate scram within approximately 13 seconds of the first release from the fuel.1/

11. Localized overheating on an even smaller scale, i.e., on the order of one to ten of the 64 fuel rods in a bundle, can be detected by PNPP's off-gas monitors.2/ The off-gas monitors detect the presence of fission products in the steam after it has passed through the turbines and is condensed. Radioactive gas released by cladding failures does not condense at the condensation temperature of steam, so that once the steam has condensed, what is left is greatly concentrated radioactive gas. The off-gas monitors would detect cladding failures approximately two minutes after they occurred. Although the radiation monitoring systems will only detect fission products if the

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1/ General Electric Co., Consequences of a Postulated Flow Blockage Accident in a Boiling Water Reactor, NEDO 10179, Rev. 1., pp. 8-2 to 8-3.

2/ Id. at 8-3.

main steamline isolation valves are open, closure of these valves will not occur without reactor scram. Moreover, localized overheating can only occur prior to scram. Thus, closure of the main steamline isolation valves does not detract from the ability of the radiation monitors to detect localized overheating. While both the off-gas monitors and the steamline radiation monitors can be used to detect localized overheating, they provide an additional method for detecting generalized ICC as well. A LOCA in which the core became uncovered would generate the fission products necessary to activate these systems, which would indicate ICC independently of PNPP's water level measurement systems.

Evaluation of PNPP's Existing Instrumentation

12. As stated in ¶ 3, above, my statements in this affidavit are based in part on three major reports prepared by S. Levy, Inc. for the BWROG. Two of these reports, Review Of BWR Reactor Vessel Water Level Measurement Systems, July 1982 (SLI-8211) and Inadequate Core Cooling Detection In Boiling Water Reactors, November 1982 (SLI-8218), provide the basis for determining whether PNPP's existing instrumentation is adequate for detecting ICC. The third report, SLI-8121, analyzes in-core thermocouples in BWRs (see ¶¶ 3 and 4 above).

13. SLI-8211 contains a detailed analysis of the water level measurement systems in various BWRs, including a BWR 6

Mark III, the design used at PNPP. The report concludes that water level measurement systems for PNPP-type BWRs are highly reliable. Only two situations were found that can cause inaccurate water level measurements. Both are associated with drywell overheating. In the first case, excessively high drywell temperatures can cause "flashing" in the reference leg, and subsequent boiloff of the water level in that leg. If the reactor vessel is depressurized, the combination of high temperature and low pressure can cause the reference leg water to flash to steam and boil off, causing the pressure sensors to indicate that coolant level in the reactor is higher than it actually is. In the second situation, excessively high drywell temperatures can cause density changes in the liquid in the portion of the piping in the drywell. Density changes in the reference or variable leg affect the pressure which the fluid in those legs exerts on the pressure sensor, and hence affect the indicated water level.

14. SLI-8211 concludes that three design changes should be implemented in PNPP-type BWRs to eliminate or minimize the effects of flashing and density changes caused by drywell overheating. First, the vertical drop of the reference leg piping in the drywell should be minimized, since the instrumentation can erroneously indicate as much as 16 additional inches of reactor coolant level per foot of vertical drop of reference leg piping, once flashing has occurred. Second, the vertical drops

of the reference leg piping and variable leg piping in the drywell should be approximately equal. Density changes in the fluid in the piping caused by drywell overheating can cause the pressure sensors to be off by as much as 12% of the difference in vertical drop between the reference and variable legs. If the vertical drop of the drywell piping is the same in the two legs, density changes which may occur affect each leg nearly equally, minimizing the error the density change could otherwise cause. Third, the flow-restricting orifices in the reference legs should be moved as close to the drywell wall as possible. If flow-restricting orifices are located close to the reactor vessel, flashing in the drywell piping causes a pressure gradient across the orifice which causes erroneous water level indications. This problem is virtually eliminated if the orifices are located in the piping where it passes through the drywell wall. The affidavits of Frank Stead and Gary Leidich describe the steps taken at PNPP to implement these recommendations. As a result of the changes described in the Stead and Leidich affidavits, the effects of drywell overheating are insignificant at PNPP.

15. SLI-8211 also considers the reliability of system trips at PNPP-type BWRs when water level measurement system failures occur. A failure analysis study was performed on the water level measurement system of a BWR 6 Mark III (PNPP-type) plant to determine plant vulnerability to potential

combinations of instrument failures and the consequences of each such combination of failures. Analyses were performed assuming a reference leg break or leak (which would cause all instruments associated with that reference leg to give erroneous readings), in conjunction with one of a number of instrument failures. Instrument failures were postulated for every type of instrument associated with water level measurement in BWR 6 Mark III plants. Each instrument failure was assumed to occur in a set of instruments other than the one in which the reference leg was assumed to have broken. After analyzing these combinations of system failures, SLI-8211 concludes that no serious consequences would result from any of the failures listed above in conjunction with an unrelated reference leg break. No challenge to fuel design limits, danger of core uncovery, or need for unusual operator action to mitigate the consequences of such failures is apparent. SLI-8211 therefore concludes that the water level measurement systems for PNPP-type BWRs will function adequately even where the combination of failures described above occurs.

16. Finally, SLI-8211 reviews available industry data of actual events involving water level measurement system failures to insure that all potential concerns have been considered. In addition to the problems identified above, industry data shows that mechanical instrumentation utilized in some BWR plants has had a relatively large number of failures. However, PNPP uses

analog, not mechanical, instrumentation in its water level measurement system. Consequently, PNPP is not vulnerable to this type of failure. SLI-8211's recommendation that mechanical instrumentation should be replaced or supplemented with analog instruments is therefore not applicable to PNPP.

17. The second report referred to in paragraph 12, SLI-8218, contains a general analysis of ICC detection in BWRs (as opposed to SLI-8211, which concentrates on the adequacy of water level measurement systems only). As stated in paragraph 3 above, in 1981 the NRC Staff had taken the position that in-core thermocouples were necessary to indicate ICC in BWRs. To determine whether additional instrumentation is necessary to detect ICC in BWRs, a general study of ICC detection in BWRs was undertaken for the BWROG, and reported in SLI-8218.

18. The study documented in SLI-8218 first examines the relationship between water level in the reactor core and the approach of, existence of, and return from ICC. The study concludes that because the geometry of the core and reactor internals permits free flow of water to the fuel bundles, and because the fuel cladding will not overheat in the presence of cooling water, water level is a conclusive indicator of the adequacy of core cooling. The study then assesses the probability of a core melt accident due to the failure of existing BWR water level measurement systems to detect ICC. The study concludes that for BWRs in general, that probability was

approximately 1.22 events per million reactor-years. The majority of this probability is due to the effects of drywell overheating; this problem, however, is insignificant at PNPP (see ¶ 14). Because most of the probability of inaccurate water level measurement is attributable to the effects of drywell overheating, the study shows that the probability of a core melt accident due to water level instrumentation failure at PNPP is approximately 0.48 events/million reactor-years. The study also surveys incore thermocouples as alternative ICC detection devices. The study analyzes possible locations for the thermocouples and evaluates their performance and problems associated with their use in these possible locations.

USE OF IN-CORE THERMOCOUPLES TO DETECT ICC

Description and Design of Thermocouples

19. As stated above, it is OCRE's contention that in-core thermocouples should be used as a means of detecting ICC by measuring the temperature of the fuel rods in the reactor core. Thermocouples are devices used to measure temperature changes in the environment in which they are located by generating small voltage charges (millivolts). A thermocouple consists of two dissimilar metals which are placed in contact with one another; the dissimilar metals are continued from the thermocouple as lead wires to a sensitive voltage measuring device. The lead wires must be electrically isolated, i.e., insulated from each other and from ground. All connections along the

leads must be to leads of similar metals and be electrically sound. In addition, a ground shield must surround the leads to prevent the small voltage charge generated by the thermocouple from being submerged in electrical noise from the outside.

20. In addition to these general requirements, thermocouples which would be installed in a BWR have certain specific requirements. The thermocouple and its insulated and shielded leads must be protected from the high temperature water and steam in the reactor. This can be done by placing them in a pressure tube which penetrates the reactor vessel and extends to the thermocouple location. The pressure tube must be structurally supported and protected to reliably withstand the physical stresses which will be imposed on it by core fluid flow, refueling operations, and maintenance operations within the reactor. This is important because any mechanical failure (i.e., breaking or loosening) of the thermocouple pressure tubes creates the potential for damaging pumps, jamming mechanisms such as control rod drives, or obstructing fluid flows, particularly in fuel assemblies. In addition, the thermocouple tubes must be installed so as not to interfere with the function and operation of the reactor. For instance, the normal nuclear reaction, heat transfer, core flow, and fuel cooling functions should not be adversely affected by the installation of thermocouple pressure tubes. Moreover, the location and housing of thermocouples must be designed to permit the removal of

internals such as driers, separators, fuel and in-core instruments during refueling, as well as permitting removal and maintenance of the thermocouples themselves. These requirements and constraints severely limit the thermocouple locations for which a reliable design can be achieved. Examination of the potential locations illustrates this conclusion.

Possible Locations of In-Core Thermocouples in BWRs

21. Position (1) - Inside Fuel Assembly: In this case, the pressure tubes containing the thermocouples and lead wiring would be attached to the fuel rod cladding (inside the fuel channel box, which contains 64 fuel rods). Thermocouples in this location would measure the surface temperature of the cladding to which they were attached. The pressure tube containing the insulated and shielded leads could run along the fuel rods or fuel channel, pass through the top of the fuel assembly, and traverse the upper core grid by penetrating the core shroud and then the wall of the reactor vessel. Alternatively, the thermocouple pressure tubes might be routed through the upper plenum head of the steam separator and pass through the reactor wall. Pressure tube and lead wire disconnects would have to be provided for each fuel assembly containing these thermocouple tubes to permit removal of the fuel. If routed through the upper plenum head, additional disconnects would be required to permit removal of the steam dryer and steam separator assemblies during refueling.

22. A reliable installation for thermocouples inside the fuel assembly is not considered possible for the following reasons. The presence of a pressure tube within the fuel assembly could detrimentally affect the thermal-hydraulics of the reactor coolant. For example, a thermocouple tube might cause local overheating of the fuel cladding, by acting as an insulator and interfering with heat transfer from the cladding to the coolant.3/ Moreover, a thermocouple attachment which is strong enough to withstand core stresses might reduce the cladding integrity, thus increasing the chance of fission product leakage. In addition, the necessary pressure tube and lead wire disconnects would be very difficult to design and install for reliable repeated connection and disconnection remotely under water. For example, it would be difficult to design electrical connections for the thermocouple pressure tubes which could be made to function under water without water entering the pressure tube. Finally, thermocouples on fuel cladding exist only in test reactors which provide operating experience that is not directly applicable to commercial reactors. Unlike commercial reactors, test reactors can be designed to accommodate thermocouples. For example, test reactors are not designed for efficient refueling operations. In addition, they have, by

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3/ F. Moyinger, "Two-Phase Flows and Heat Transfer with Application to Nuclear Reactor Design Problems" (J. Gignoux, ed.), von Karman Institute for Fluid Dynamics, Rhode-Saint-Genese, Belgium, pp. 375-381.

comparison to commercial reactors, few fuel bundles and consequently fewer thermocouples, pressure tubes, disconnects and vessel penetrations to install and maintain. Test reactors also operate for shorter time periods than commercial reactors, which significantly affects the designed life of the thermocouples. The reliability of thermocouples inside the fuel assemblies of commercial power reactors is untested.

23. Position (2) - At Exit of Fuel Assembly:4/ In this case, thermocouples would be located at the top of various fuel bundles and would measure the exit temperature of the coolant at this location. In order to place thermocouples in this location (without attaching the thermocouple tubes to the fuel cladding) a structure would have to be installed over the core, below the upper plenum head of the steam separators, to suspend the thermocouples in their pressure tubes. This structure would most likely be attached to the upper plenum head and hence would require pressure tube and wire disconnects to

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4/ This is technically not an in-core location, which is the issue that OCRE has raised in its contention. A discussion of this location is included because it is close to the core, and if it were possible to place thermocouples in this location, they could distinguish localized overheating from generalized ICC. Two other ex-core locations for thermocouples inside the reactor might also be considered, in the upper plenum head of the steam separator assembly, and in the steam dome above the steam drier assembly. While a reliable installation for thermocouples is possible in the steam dome, this location is useless for detecting localized overheating, and thermocouples there are subject to the same problems discussed in ¶¶ 25-28, below. A reliable installation is not considered possible for thermocouples in the upper plenum head.

permit removal of the steam separators and dryers during refueling. A reliable installation is not considered possible for this location, because the required supporting structure would interfere with core spray distribution and thus would seriously reduce the effectiveness of one of the important emergency core cooling systems. This is so because a massive structure would be necessary to span the distance between the dome and the core and reliably resist the considerable forces of the core exit flow passing it. The problem of achieving reliable disconnects, as discussed for Position (1), above, would also be an impediment to placing thermocouples at this location.

24. Position (3) - Inside Local Power Range Monitors ("LPRM") Tubes: In this arrangement, thermocouples would be installed inside the LPRM tubes and would measure the inside temperature of the tube at that location. LPRM tubes penetrate the bottom of the reactor vessel and project vertically up into the core, passing through the space between the corner of four fuel channel boxes. Because the LPRM tubes are pressure tubes, new pressure tubes for the thermocouples and their leads would not be necessary. Examination of the LPRM tube design shows that there is space for a limited number of thermocouples. However, new end fittings for the LPRM tubes would have to be designed to accomodate the lead wires. A reliable installation is considered possible for this location because LPRM tubes and

instruments within them have operated successfully in BWRs for many years.

Problems Associated With the Use of Thermocouples

25. Paragraphs 21-24 demonstrate that there is only one feasible in-core location for BWR thermocouples being used to detect ICC: in the LPRM tubes inside the core. However, even thermocouples in the LPRM tubes are susceptible to a number of problems. The extensive analysis in SLI-8121 and SLI-8218 demonstrates that in-core thermocouples suffer at least three major drawbacks. The first is that thermocouples react to and indicate ICC due to loss of coolant only after serious time delays. This is primarily because the core must uncover, (i.e., water level must drop below the top of active fuel, leaving the fuel rods partially exposed) and considerable overheating must occur before thermocouples will react sufficiently to discriminate overheating from normal temperature variations.

26. The analysis performed in SLI-8218 shows that in the case of a relatively slow loss of cooling water, thermocouples placed in LPRM tubes near the top of the core would not indicate ICC until the collapsed water level had fallen approximately one-half of the way through the core.5/ This is a

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5/ In the case of a rapid loss of cooling water, thermocouples would be of little use to an operator because ICC would have occurred by the time thermocouples indicated overheating (see ¶ 32).

serious limitation of thermocouples for two reasons. First, for relatively slow LOCAs, thermocouples would not begin to indicate ICC until approximately 52 minutes after the water level first started to drop below normal, virtually eliminating their ability to to detect the approach of ICC. Second, over 20 feet of cooling water would be lost before thermocouples could indicate ICC. This limitation is intrinsic with thermocouples because the thermocouples do not react until there has been significant fuel overheating.

27. The second major drawback of thermocouples is that they can generate ambiguous or confusing instrumentation readings during ICC. Actuation of core sprays or opening of pressure relief valves is likely to rewet the thermocouples and thus generate inaccurate readings; that is, they would indicate normal temperatures when in fact the core is overheated. Recently, the potential for thermocouple ambiguities was illustrated at the Loss-of-Fluid Test (LOFT) facility in Idaho. After an intentional overheating of the test reactor by removal of coolant, thermocouples located one inch above the top of the core indicated that the core was recovered when in fact it was not. This occurred as a result of water droplets on the thermocouples caused by drainage from the upper plenum.6/

6/ Adams, J.P. and G.E. McCreery, "Limitations of Detecting Inadequate Core Cooling with Core Exit Thermocouples", Transactions of the ANS, vol. 46, pp. 474-476, June 1984.

Similarly, during small break LOCAs, the pressure inside the reactor vessel is not likely to be constant. Pressure in the vessel will rise to the pressure relief valve set point, and then drop when the valves open. During this pressure drop, voids will form in the reactor coolant, causing the coolant level to rise eight to twelve feet. This will periodically cover and uncover the thermocouples, causing them to indicate lower temperatures than actually exist in the core.

28. The third major drawback of thermocouples relates to their use in detecting localized overheating. This weakness of thermocouples has two aspects. First, thermocouples in the LPRM tubes (which are located between fuel channel boxes) would not indicate localized overheating as long as they were covered with water, even if the flow of water into a fuel channel box was blocked. On the other hand, the presence of additional thermocouple tubes inside the fuel channels could interfere with heat transfer/coolant flow. Thus, in-core thermocouples could actually contribute to localized overheating. Second, an excessive number of thermocouples and additional pressure tubes would be required in order to detect localized overheating reliably. Since thermocouples in an LPRM tube will not detect localized overheating caused by flow blockage in a fuel channel box, pressure tubes to house thermocouples would have to be located inside the fuel channel boxes. The PNPP reactor core has 748 channels containing fuel bundles; to be assured of

detecting localized overheating, 748 thermocouples and pressure tubes would be required, without allowing for redundancy and assuming only one thermocouple per fuel bundle, which could measure temperature at only one point along the fuel bundle. These additional pressure tubes would not only interfere with coolant flow, but would also require numerous additional penetrations of the reactor vessel.

29. As stated in paragraph 18, the likelihood that a core melt accident will occur at PNPP due to failure of PNPP's present water level instrumentation is approximately 0.48 events/million reactor-years. SLI-8218 found that in the event that an unambiguous ICC detection device could be designed, such a device could reduce the likelihood of a core melt accident caused by a failure to detect ICC to approximately 0.14 events per million reactor-years. Since thermocouples are <u>not</u> unambiguous in all situations (see ¶ 27), the possible benefit from the addition of thermocouples at PNPP might be a reduction in the likelihood of a core melt accident from 0.48 events/ million reactor-years to something greater than 0.14 events/ million reactor-years. Moreover, as described above, thermocouples have other drawbacks which further reduce their usefulness, and may in fact increase the likelihood of fuel damage, thereby offsetting some portion of the benefit which might be gained from their installation. Thus, the benefit, if any, which can be expected from the installation of in-core

thermocouples at PNPP is exceedingly small. The conclusion which can be drawn from SLI-8218 is that the installation of an alternative ICC detection device such as thermocouples is not warranted at PNPP.

ANALYSIS OF STATED BASES FOR OCRE'S CONTENTION

30. As stated in ¶ 4, OCRE contends that in-core thermocouples should be used at PNPP as an alternative means of detecting ICC. The first basis for this contention is "the requirement of Regulatory Guide 1.97, Revision 2 and TMI Action Plan item II.F.2." At the time the NRC Staff issued these documents (Nov.-Dec. 1980), the Staff took the position that in-core thermocouples should be used in BWRs to detect ICC. The NRC Staff has subsequently modified its position, however, and no longer requires in-core thermocouples. The TMI Action Plan (NUREG-0737), for example, was modified by Supplement 1 to NUREG-0737, issued in December, 1982. Section 6.1(c)(i) of the supplement states:

> BWR in-core thermocouples . . . are not required pending their further development and consideration as requirements.

Similarly, Reg. Guide 1.97, Rev. 2 (Dec., 1980) has been superseded by Reg. Guide 1.97, Rev. 3 (May, 1983). The NRC Staff's position, as stated in Rev. 2., called for "four thermocouples per quadrant. A minimum of one measurement per quadrant is

required for operation." By contrast, Rev. 3 does not mention in-core thermocouples, stating only that monitoring core cooling through measurements other than water level (e.g., thermocouples) is "still being considered, subject to further development."7/ Consequently, it is clear that the NRC staff no longer considers in-core thermocouples to be required.

31. The second basis OCRE gives for its contention is an analysis performed by Battelle Laboratories which investigated thermocouple response time in detecting ICC. The Battelle report concurs with the analysis of SLI-8121 in reporting that water level must sink well into the core before ICC can occur and that thermocouples must be uncovered before they will begin to detect and respond to ICC. The implications of this are twofold. First, water level measurement systems, unlike thermocouples, have the entire time that water level is dropping down to and into the reactor core to detect and trigger

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7/ Recently the NRC Staff has further modified and clarified its position with respect to NUREG 0737, Item II.F.2. In Generic Letter No. 84-23 (Oct. 26, 1984), the Staff indicates two categories of improvements to water level measurement systems which it deems necessary in BWRs: improvements that will reduce level indication errors caused by high drywell temperatures; and replacing mechanical level indication equipment with analog equipment. The Staff has stated that implementing these improvements "will give increased assurance that the level instrumentation will provide the inadequate core cooling instrumentation required by NUREG 0737, Item II.F.2 and thereby satisfy this requirement." As described in ¶¶ 14 and 17, above, PNPP has already implemented changes in its water level measurement system to minimize the effects of drywell overheating, and uses analog, rather than mechanical, level indication equipment.

plant responses to ICC. As stated in paragraph 26, above, during relatively slow LOCAs, water level would be dropping for approximately 52 minutes before in-core thermocouples would indicate ICC. Second, the core would be significantly overheated by the time the thermocouples provided an indication of overheating. Therefore, the Battelle report supports the conclusion that thermocouples would provide little or no useful warning during a relatively slow LOCA.

32. Part of the analysis performed in SLI-8121 included a review of the Battelle analysis and a comparison of the Battelle findings with those of S. Levy, Inc. In summary, the comparison shows that the more rapid response reported by Battelle (1.5 minutes after the thermocouple is uncovered vs. 13 minutes as calculated by SLI) is traceable to (1) the high decay heat Battelle assumed and (2) the assumed absence of convective steam cooling. Both of these assumptions significantly reduce the response times for thermocouples. The S. Levy, Inc. calculation is applicable to accidents involving a relatively slow loss of cooling water during which thermocouples might give the operator time to prevent further overheating. The Battelle analysis, on the other hand, is based on assumptions which only apply in cases of extremely rapid loss of inventory which occur so quickly that remedial action by the emergency core cooling system must be automatically initiated. Thus there is little or no usefulness to the operator in having

thermocouple detection for relatively rapid LOCAs. The more rapid response calculated in the Battelle analysis therefore does not support the installation of in-core thermocouples at PNPP.

33. The third basis OCRE gives for its contention is a discussion by Dr. Richard E. Webb in The Accident Hazards of Nuclear Power Plants (hereinafter Webb), of a report prepared by General Electric, "Consequences of a Postulated Flow Blockage Incident in a Boiling Water Reactor" (May 1970). OCRE has theorized, with reference to pp. 59-61 of Dr. Webb's book, that blockage of coolant flow around a fuel bundle could cause localized overheating, which in turn would cause the fuel and cladding to melt, overheat neighboring fuel bundles, and propagate a meltdown throughout the core. OCRE maintains that in-core thermocouples (presumably 748, or one per fuel bundle) are necessary at PNPP because they can detect localized overheating before fuel failure, thereby avoiding a propagating core melt accident or steam explosions. Of course, thermocouples used for this purpose are subject to the problems discussed above (see ¶ 28), including slow response time, excessive number, contribution to localized overheating, etc. Moreover, the G.E. Report concluded, based on extensive review of the applicable technology, analyses, and supplementary tests, that localized overheating would not lead to steam explosions or melting propagation throughout the core.

34. Experimental work conducted in recent years reinforces GE's original conclusions. The melting temperature of the oxidized fuel and cladding is well above the temperature at which the metallic channel box would melt and permit water to enter the bundle. If a scram is assumed to have occurred so that the heat generated by the reactor is relatively low (i.e. decay heat), radiation heat transfer would assure that the cladding and fuel temperature would not be significantly above the channel box temperature. When the water entered, it would interact with materials that are well below the temperature at which steam explosions can be supported.8/ Analysis of the TMI-2 accident by Garry R. Thomas9/ confirms the conclusion of the GE report that propagation will not occur in the presence of water. At TMI-2, one-third to one-half of the entire core was reduced to rubble by overheating. Nevertheless, the events at TMI-2 show that the introduction of water will cool the core and that propagation does not occur in the presence of

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8/ Lloyd S. Nelson, "Steam Explosion Studies With Single Drops of Corium-Related Melts: Ferrous and U- and Zr- containing Oxides," Proceedings of the International Meeting on Light Water Reactor Severe Accident Evaluation, August 28, to September 1, 1983, 6.7-1 to 6.7-3. The tests reported by Nelson were designed and conducted to promote steam explosions, and they showed that fuel, cladding and channel materials will only support a steam explosion if they are highly oxidized and molten. Because the oxidized materials would not be molten when water entered, steam explosions could not occur.

9/ "Overview-Significant Aspects of TMI-2 Core Material Performance," Transactions of the ANS, vol. 35, pp. 194 to 196, (November, 1980)

water.10/

35. OCRE has raised one other argument in favor of thermocouples in its Motion For Leave To File Contention 14, dated August 18, 1982. OCRE cites a G.E. study, General Electric Evaluation of the Need For BWR Core Thermocouples (October, 1981), as stating that thermocouples will provide useful and unambiguous information in the event of a loss of cooling water (uncovering the core) with no normal, emergency, or makeup water systems available to replenish cooling water. In the extraordinarily unlikely event of the combination of system failures that OCRE has postulated, the core would be cooled by water and steam flow for a considerable period of time until the water in the core region has boiled off. The G.E. report states that under such conditions, thermocouples above the core could provide unambiguous information by indicating ICC. The G.E. report explicitly limits this statement to thermocouples above the core and thus provides no

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10/ Dr. Webb was also critical of GE's analysis of the accident progression if a scram was assumed not to occur. However, this would require failure of the redundant steamline radiation monitoring system. Thermocouples would only be of value in this highly unlikely situation to provide indication of overheating to the operator. However, this function would be performed by the off-gas monitoring system. Moreover, considerable work has been done to date which indicates that the potential for steam explosions, even if no scram is assumed, is considered to be much less than it was at the time of Webb's writing. See, e.g., M.C. Leverett and D. Squarer, "Steam Explosions in Perspective", Proceedings of the International Meeting on Light Water Reactor Severe Accident Evaluation, August 28 to September 1, 1983, 6.1.

support for the proposition that in-core thermocouples would provide unambiguous information in this situation. Moreover, should this condition occur, the operator would be taking appropriate steps to replenish water and/or scram the reactor based on indications of low water level provided by PNPP's water level measurement systems and radiation monitoring systems.

CONCLUSION

36. Existing water level measurement systems at PNPP provide an accurate and reliable means to detect and respond to various stages of ICC. By contrast, a reliable and practical thermocouple installation in PNPP would be difficult to achieve and would have very limited usefulness. Thermocouples would have to be placed inside the LPRM tubes currently located in the PNPP core. Thermocouples in this location can generate ambiguous readings during ICC. In addition, thermocouples are subject to serious time delays in detecting ICC, virtually eliminating their ability to detect the approach of ICC. Moreover, the addition of in-core thermocouples could degrade the structural integrity of the fuel cladding and/or contribute to localized overheating. PNPP's existing water level monitoring system and radiation monitoring systems are reliable, redundant and unambiguous in detecting the approach and existence of ICC. Thus, the installation of thermocouples at PNPP would not significantly reduce the already small likelihood that a core melt accident would occur at PNPP due to the failure to detect ICC.

In sum, a thorough investigation of thermocouples leads to the conclusion that their installation at PNPP is not warranted.

Charles B. Johnson

Subscribed and sworn to
before me this 11th day
of January, 1985.

OFFICIAL SEAL
JAMES H. DUGAN
NOTARY PUBLIC - CALIFORNIA
SANTA CLARA COUNTY
My Comm. Expires Dec. 20, 1985

Notary Public

EXHIBIT A

INDIVIDUAL EXPERIENCE RECORD

NAME: Charles B. Johnson

BUSINESS ADDRESS: S. Levy, Incorporated
3425 South Bascom Avenue
Campbell, California 95008
Telephone: 408-377-4870

EDUCATION: Stanford University - Palo Alto, California
Bachelor of Science Degree in Mechanical Engineering, June 1958.

LICENSES: Licensed Professional Engineer (Mechanical)
State of California

WORK EXPERIENCE:

7/80 to Present — S. Levy, Incorporated, Campbell, California. General Manager and Consultant.

Led and contributed to studies related to water level measurement in Boiling Water Reactors. Results have been presented to and accepted by Nuclear Regulatory Commission staff.

Led and contributed to studies related to reactor safety. These include inadequate core cooling detection and prevention and mitigation of core meltdowns.

General Electric Company, Nuclear Energy Division

1976 to 1980 — Engineering Manager - Valves and Auxiliary Equipment
Responsible for technical procurement of nuclear power plant components.

Designed and manufactured nuclear equipment (steam flow restrictors, pipe whip restraints).

1973 to 1976 — Unit Engineering Manager - Pressure Regulation Systems
Managed unit designing and integrating a pressure regulation system into product scope.

1969 to 1972	Unit Engineering Manager - Advanced Design Conceived and evaluated product improvements for boiling water nuclear reactors.
1966 to 1969	Engineer - Auxiliary System Design Designed emergency core cooling complex for nuclear reactors.
1964 to 1966	Engineer - Performance Analysis Predicted course of loss of coolant accidents in boiling water nuclear reactors.
1961 to 1964	Program Engineer Rotated among different departments in the plant and other GE locations as part of company's training and development program.
1958 to 1961	U.S. Air Force Entered active duty from ROTC into the U.S. Air Force in 8/58. Completed one year training in weather forecasting at New York University. Served as weather forecaster in the United States. Honorably separated from service in 8/61 as 1st lieutenant.

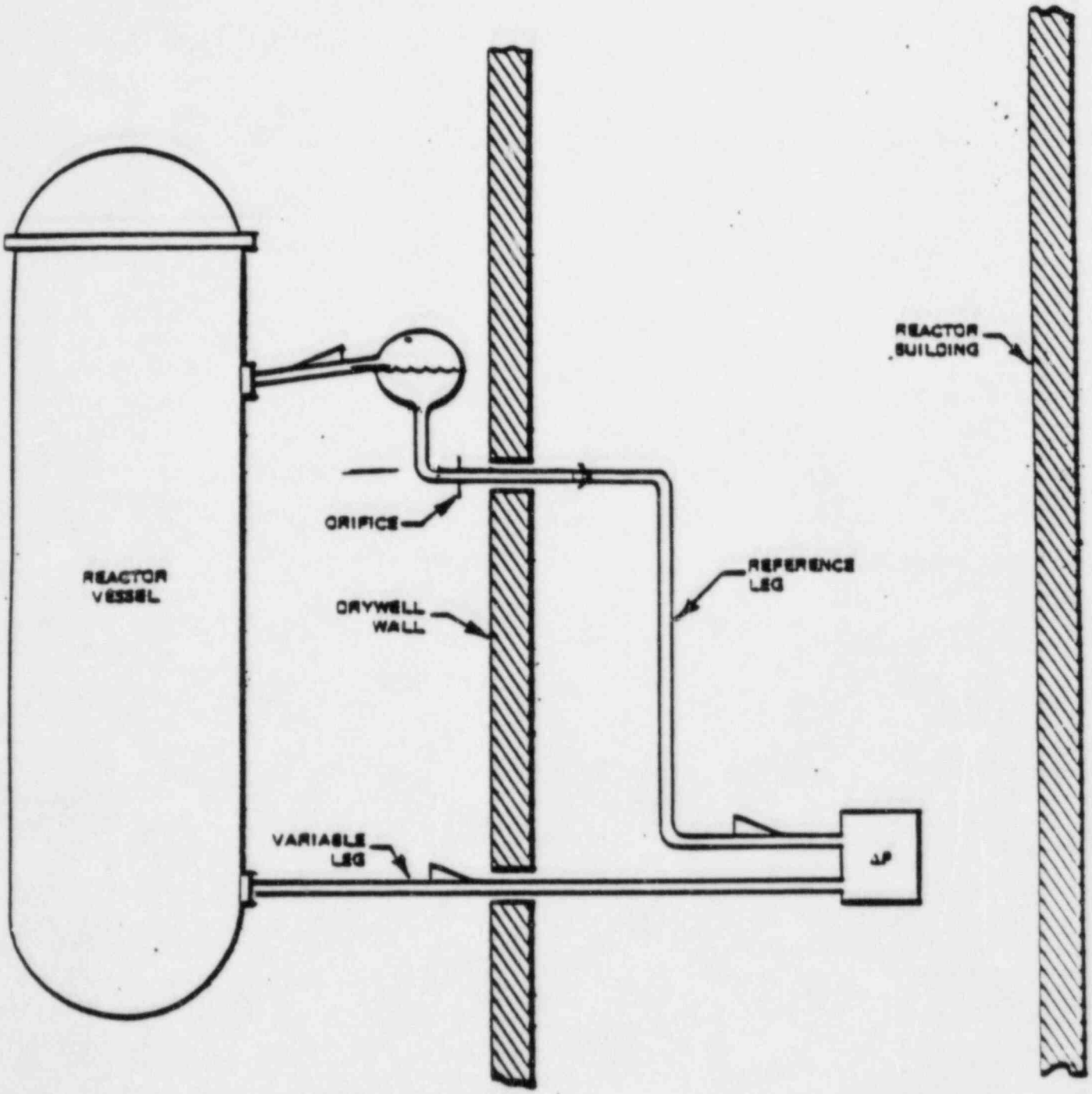
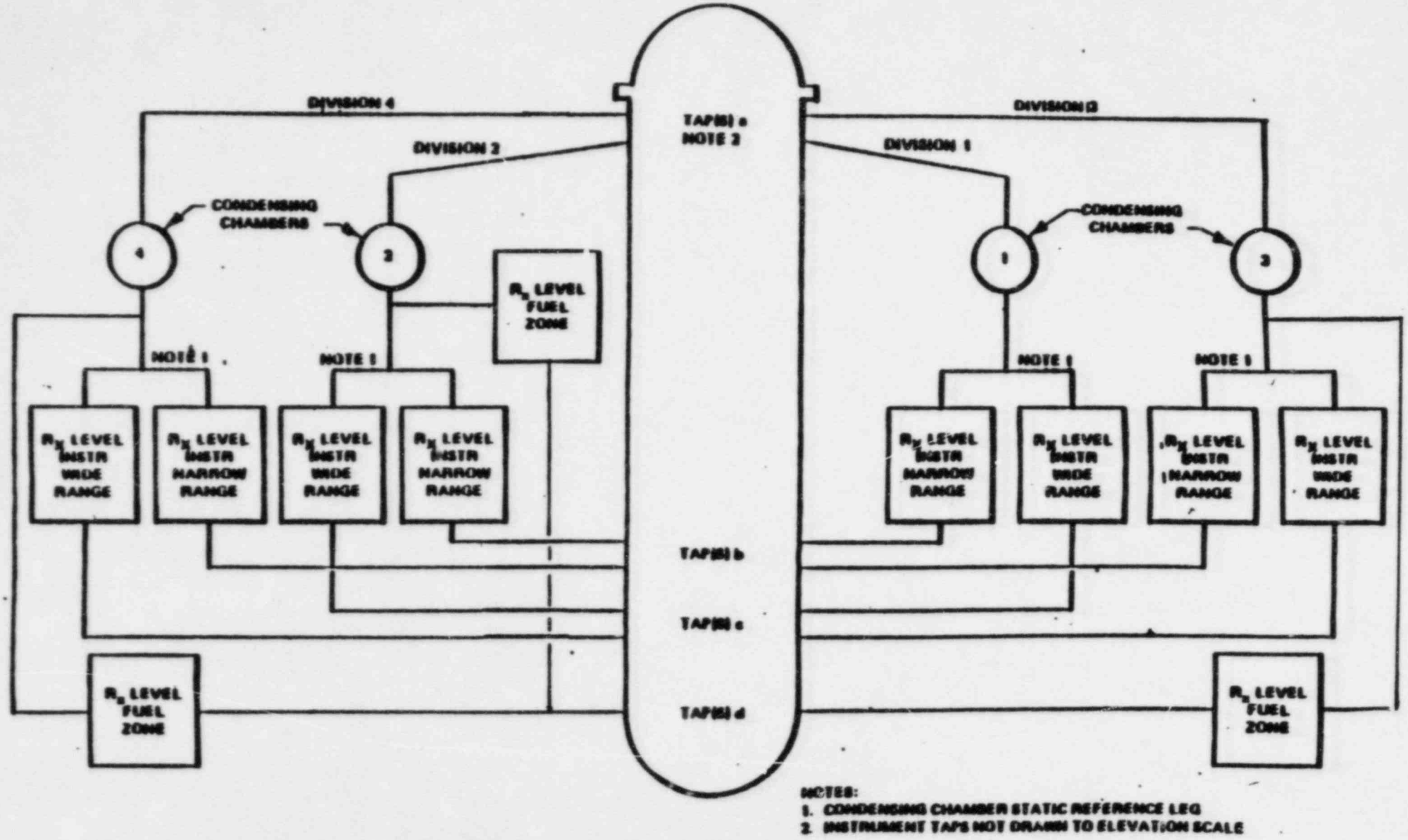

REACTOR BUILDING
ORIFICE
REFERENCE LEG
REACTOR VESSEL
DRYWELL WALL
VARIABLE LEG

Figure 1



PNPP (BWR 6 MARK III) Reactor Vessel Level Instrumentation Schematic

Figure 2