UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

Before the Atomic Safety and Licensing Board

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| In the Matter of | ) | |
| | ) | |
| Philadelphia Electric Company | ) | Docket Nos. 50-352 |
| | ) | 50-353 |
| (Limerick Generating Station, | ) | |
| Units 1 and 2) | ) | |

# APPLICANT'S MEMORANDUM IN SUPPORT OF ITS MOTION FOR SUMMARY DISPOSITION OF CONTENTION I-62

## Legal Introduction

The legal analysis found at pages 1 through 6 of "Applicant's Memorandum in Support of its Motion for Summary Disposition of Contention V-4" is incorporated herein by reference.[1/]

## Argument

In its Special Prehearing Conference Order ("SPCO"), the Atomic Safety and Licensing Board ("Licensing Board" or "Board") admitted the following contention, submitted by Mr. Marvin I. Lewis and renumbered as Contention I-62:

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1/ Applicant's Memorandum in Support of its Motion for Summary Disposition of Contention V-4 (September 27, 1983).

> The Limerick nuclear power plant can suffer a major breach of containment due to pressurized thermal shock.2/

Pursuant to the SPCO, the Applicant directed requests for document production and a number of interrogatories to Mr. Lewis.3/ In his response dated June 15, 1983, Mr. Lewis stated, inter alia:

> I do not depend on any document as a basis that PTS is a problem in BWRS. Conversely, I do depend upon many documents to show in the negative that PTS has not been considered a problem in BWRs and therefore has not been investigated sufficiently in BWRs. . . .

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2/ Philadelphia Electric Company (Limerick Generating Station, Units 1 and 2), LBP-82-43A, 15 NRC 1423, 1508 (1982). In a motion dated September 28, 1983, Mr. Lewis moved that, due to material disclosed during discovery, the Board reword Contention I-62 as follows:

> Staff's dependence upon a comparison of Limerick to PWRs in which PTS has been studied in order to develop its low probability for PTS failure at Limerick, is not justified. An operating license should not be issued without a full engineering analysis of PTS at Limerick, since PTS can cause a major breach of containment, which is an accident beyond the design basis.

Motion to Reword Contention I-62 In Response to the Board's M&O of September 13, 1983 (September 28, 1983). On October 6, 1983, Applicant filed an answer opposing Mr. Lewis' motion on the ground that Mr. Lewis did not comply with the Board's September 13 Memorandum and Order by narrowing the scope of the contention. Applicant's Answer to Motions to Reword Contentions I-62 and V-4 (October 6, 1983). The Board has not ruled on Mr. Lewis' motion.

3/ Applicant's First Set of Interrogatories and Request for Production of Documents to Marvin I. Lewis (June 3, 1983).

> This contention is not based upon any study or calculation. . . . Generally, I assert that not enough or no actual numerical calculations have been made to determine that PTS is not a danger to integrity of BWR's. In other words, my basis for my contention is a lack of basis for the NRC and Licensee's answers that PTS is not a problem in BWRs.4/

The Statement of Sampath Ranganath, Lloyd S. Burns, Jr., Franklin E. Cooke, and Stephen E. Carter in Support of Motion for Summary Disposition of Contention I-62 ("Statement"), which is attached hereto and incorporated herein by reference, responds to the contention and demonstrates that it has no basis.5/

This contention was reviewed and analyzed by four employees of the Nuclear Engineering Division of the General Electric Company ("GE"), the supplier of the nuclear steam supply system including the reactor pressure vessel, specifically Sampath Ranganath, Lloyd S. Burns, Jr., Franklin E. Cooke, and Stephen E. Carter. Mr. Ranganath is the Manager of the Mechanics Analysis group in GE's Nuclear Engineering Division (Statement at ¶A.1a). He received a doctoral degree in engineering from Brown University in 1971 and is responsible for all analytical work on the Limerick

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4/ Answer to Applicant's First Set of Interrogatories to Marvin I. Lewis (June 15, 1983).

5/ This Statement is supported by the individual affidavits of Sampath Ranganath, Lloyd S. Burns, Jr., Franklin E. Cooke, and Stephen E. Carter which specify their particular participation.

Generating Station in the areas of stress analysis, fracture mechanics, fatigue evaluations, finite element methods development, stress corrosion cracking, residual stress analysis, and dynamic margin of components (Id.). Of particular importance here is the fact that Mr. Ranganath has been a key contributor to several EPRI programs relating to improved fatigue design rules for carbon steel and fracture mechanics evaluation of the pressure vessel under LOCA conditions (Statement of Professional Qualifications of Sampath Ranganath at 1). As a member of the Subgroup on Standards and Evaluation, Section XI, ASME Code, he has contributed to the development of rules for evaluating flaws in nuclear pressure vessel components (Id. at 2). Mr. Ranganath is also an adjunct lecturer at the University of Santa Clara, teaching courses in fracture mechanics and pressure vessel design (Id.). Mr. Ranganath is clearly an expert in the fields of stress analysis and fracture mechanics.

The second affiant is Lloyd S. Burns who is a Senior Engineer and Technical Leader of the Containment and Radiological Engineering group of GE's Nuclear Engineering Division (Statement at ¶A.1b). He has a Bachelor of Arts degree in Physics from Kalamazoo College and is responsible for the calculation of neutron flux and fluence on BWR reactor pressure vessels including those at the Limerick Generating Station (Id.). Mr. Burns has been employed in

the area of radiation analysis by GE since 1956 and is clearly an expert in the area.

The third affiant is Franklin E. Cooke who is a Principal Design Engineer with the Reactor Pressure Vessel and Internals Design group of GE's Nuclear Engineering Division (Statement at ¶A.1c). Mr. Cooke has a Bachelor of Science degree in Mechanical Engineering from Southern Methodist University and is employed in the area of nuclear steam supply system design, testing, and operation (Id.). More specifically, Mr. Cooke has been working in the area of fracture toughness requirements since 1974 (Statement of Professional Qualifications of Franklin E. Cooke at 1). He is responsible for the definition of fracture toughness operating limits for the reactor vessel, the definition of standard plant reactor specifications, review of product safety standards and regulatory guides, thermal cycle evaluations, and the definition of reactor water level operating limits (Id. at 2). Mr. Cooke is clearly an expert in these matters.

The fourth affiant is Stephen E. Carter who is an engineer in the Plant Materials Application group of GE's Nuclear Engineering Division (Statement at ¶A.1d). He has a Bachelor of Science degree in Metallurgy from Pennsylvania State University and has worked in the area of design, procurement, and quality control of the reactor pressure vessel and piping (Id.). In that capacity, Mr. Carter is responsible for assuring compliance with the requirements of

10 C.F.R. Part 50, Appendices G and H (Id.). Copies of the complete statement of the professional qualifications of each of these affiants are attached to the Statement and are incorporated herein by reference.

Messrs. Ranganath, Burns, Cooke, and Carter have analyzed the Lewis Contention I-62. Their analysis is based on their collective extensive experience regarding the subject matter. As discussed in more detail below, these experts concluded that the design, construction, testing, operation, and surveillance of the Limerick Generating Station, when combined with the physical behavior of a BWR, assure that pressurized thermal shock is not a problem for the Limerick Generating Station (Statement at ¶A.5).

A proper analysis of this contention must start with the definition of pressurized thermal shock ("PTS"). Pressurized thermal shock is a condition that results from the introduction of cold water into a hot pressure vessel while the pressure is high (Statement at ¶A.4). Thermal stresses are produced in the vessel walls when cold water is introduced into the vessel (Id.). Similarly, stresses are produced when the vessel is subjected to high pressures (Id.). These thermal stresses, when combined with the stresses which occur as a result of high vessel pressure, have the potential to cause crack propagation in vessel materials (Id.). Relatedly, the materials of which the reactor pressure vessel is made can become embrittled as a result of substantial neutron bombardment (Id.). If the

vessel materials are embrittled significantly due to radiation effects, this embrittlement could, under certain conditions, adversely affect the ability of this material to withstand these combined stresses (Id.). The phenomenon of PTS has been recognized as a problem in some pressurized water reactors because (1) pressure for some events can remain high in a PWR during cold water injections and (2) the neutron flux is high enough to cause significant vessel material embrittlement (Id.).

The prerequisites of PTS, high reactor vessel pressure during cold water injection and significant neutron irradiation embrittlement, do not occur in a BWR (Statement at ¶A.5). Furthermore, the decrease in vessel material fracture toughness as a result of irradiation is substantially less in a BWR than that in a PWR (Id.). Specifically, the pressure in a BWR follows the water-steam saturation curve (Id.). This means that during cold water injection, such as from the High Pressure Coolant Injection ("HPCI") system, the pressure in the BWR drops because water and steam remain in equilibrium (Id.). Additionally, the neutron fluence at the vessel wall in a BWR is very low compared with a PWR due to the presence of a large water-filled annulus between the vessel and the shroud surrounding the reactor core and due to a substantially lower reactor core power density (Id.). The consequence of these two factors is minimal radiation embrittlement of BWR vessel materials (Id.). Thus, the physical behavior of a

BWR, as stated above, assures that PTS is not a problem for the Limerick Generating Station (Id.).

The design and construction of a BWR also assure that PTS is not a problem. The Limerick reactor pressure vessels[6/] were designed, fabricated, tested, inspected, and stamped in accordance with the requirements of ASME Code Section III, Nuclear Power Plant Components, Class 1, including the Summer 1969 Addenda and ASME Code Section IX, Welding Specifications, including Summer 1969 Addenda (Statement at ¶A.6). The fabrication of the reactor pressure vessels was also performed in accordance with GE approved drawings, fabrication procedures, and test procedures (Statement at ¶A.7). Each shell and vessel head was made from formed low-alloy steel plates, and the flanges and nozzles were made from low-alloy steel forgings (Id.).

With regard to welding, these vessel components were welded in accordance with ASME Sections III and IX requirements (Id.). Required weld test samples were performed for each procedure for major vessel full penetration welds (Id.). Additionally, all plate, forgings, and bolting were 100% ultrasonically tested and surface examined by magnetic particle methods or liquid penetrant methods in accordance with ASME Code III requirements (Id.).

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6/ The Limerick reactor pressure vessels are vertical, cylindrical pressure vessels of welded construction. Statement at ¶A.7.

The pressure retaining welds were also ultrasonically examined in accordance with ASME Code Section XI and Regulatory Guide 1.150 (Rev. 1) guidelines (Id.).

Fracture toughness, a material's inherent ability to resist unstable extension of flaws in the presence of applied, dynamic, impact, or other suddenly applied loads, was measured and controlled in accordance with ASME Code Section III requirements to limits specified by GE (Statement at ¶¶ A.7 and A.8). Appendix G of 10 C.F.R. Part 50 specifies minimum fracture toughness requirements for the ferritic materials in pressure-retaining components of the reactor coolant pressure boundary of water cooled power reactors in order to provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests, to which the pressure boundary may be subjected over its service lifetime (Statement at ¶A.8).

Section 5.3.1.5 of the Final Safety Analysis Report ("FSAR") demonstrates the compliance of the Limerick reactor vessel design with the requirements of 10 C.F.R. Part 50 Appendix G (Id.). As described there, the ferritic pressure boundary materials of the Limerick Unit 1 reactor pressure vessel were qualified by toughness testing in accordance with the 1968 edition of the ASME Code including the Summer

1969 Addenda (Id.).[7/] In addition, these materials were tested to the augmented fracture toughness requirements specified by GE (Id.). All reactor pressure vessel components were impact tested by either the drop-weight test or the Charpy V-Notch Impact test (Id.). Both impact tests were conducted on beltline plate material, closure flange material, top head material, feedwater and LPCI nozzle material forgings (Id.). Reference temperature nil ductility transition temperature ($RT_{NDT}$) values for the reactor pressure vessel components were established using impact test data and procedures that meet the requirements of 10 C.F.R. Part 50 Appendix G (Id.).

The fracture toughness of a material can be affected by neutron fluence (Statement at ¶A.9). The effect of neutron fluence over the life of the facility on the pressure vessel materials is to cause a decrease in the fracture toughness of the materials (Id.). This decrease in toughness, however, is significant only at high fluences, i.e., well above those expected in the Limerick vessels based upon conservative calculational techniques as verified by operational experience (Id.). A description of how the fluences were determined for the Limerick vessels is set out below.

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7/ Similarly, compliance of the Limerick Unit 2 reactor with 10 C.F.R. Part 50, Appendix G will be demonstrated (Footnote Continued)

The neutron fluence calculations were carried out on the basis of analytical models incorporating reactor core data, geometric arrangement, and basic physical data (Statement at ¶A.10). These items are incorporated into computer programs to solve the transport equations for nuclear radiation (Id.). These programs were verified in compliance with the quality assurance provisions of 10 C.F.R. Part 50 Appendix B (Id.). The results of these calculations have been compared to field measurements and have been found to overpredict conservatively the neutron fluence (Id.).

The BWR vessel geometry from the viewpoint of neutron vessel fluence calculations can be described in terms of a cross sectional view of concentric cylinders (Id.). The inner core region consists of 764 square fuel bundles of identical geometrical size arranged in a pattern simulating a cylinder (Id.). This region is surrounded by an annulus of water of an average thickness of approximately 8 inches (Id.). The core water is separated from the downcommer water by a stainless steel metal shroud 2 inches thick (Id.). The downcommer region containing the jet pumps is a water region 22.1 inches thick (Id.). The vessel is

(Footnote Continued)
prior to issuance of an operating license for that unit. Statement at ¶A.8.

approximately 6.2 inches thick in the beltline region (Id.).[8/]

A maximum fluence for the core beltline material of $1.1 \times 10^{18}$ n/cm$^2$ at one quarter of the vessel wall thickness from the inside diameter was conservatively calculated (Statement at ¶A.11). This fluence was applied over the entire length of the core beltline plates and welds in order to determine the shift in fracture toughness which was, in turn, used to determine the reactor operating limits at the end of reactor service life (Id.). For critical nozzles such as the LPCI nozzle, the fluence was uniquely calculated for each nozzle (Id.). The resulting core beltline operating limitations are less restrictive than those operating limitations established as a result of analyses of other reactor vessel parts (Id.). Such vessel parts are located well away from the core beltline in a region of insignificant fluence with respect to fracture toughness properties of the vessel material (Id.). The results of these calculations are depicted in FSAR Figure 5.3-4, a copy of which is attached to the Statement (Id.). This figure demonstrates that the core region beltline is not limiting with regard to the setting of temperature and pressure limits for the reactor (Id.). This figure will become part of the Technical Specifications for Limerick Unit 1 and will

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8/ See FSAR Figure 4.3-29.

define the minimum temperature versus reactor pressure required to assure adequate fracture toughness (Id.).[9/]

A fracture mechanics evaluation of the effects of a potential loss of coolant accident (LOCA) in a BWR/6 reactor vessel of similar design to Limerick has been performed (Statement at ¶A.12). The purpose of that analysis was to determine if crack propogation could occur as a result of stresses induced in the vessel by the injection of cold water by the Emergency Core Cooling System (ECCS) (Id.). The analysis included determination of thermal stresses in the vessel and calculation of resulting applied stress intensity factors which would exist at the tip of a potential crack in the vessel wall (Id.). The stress intensity factors were compared to the available materials fracture toughness which was calculated considering vessel temperature and neutron fluence effects on vessel material properties (Id.). It was found that the available vessel material fracture toughness always exceeded the applied stress intensity factors for all postulated crack depths (Id.). It was therefore concluded that crack propogation would not occur, even for large initial flaws with depths approaching the vessel wall thickness (Id.). Limerick specific fluence, material properties, and NDT shift data

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9/ A similar curve will be developed for Unit 2.

confirm the results of the fracture mechanics evaluation as applicable to the Limerick reactor pressure vessels (Id.).

Finally, the surveillance program for the Limerick vessels will assure that PTS is not a problem. Surveillance specimens were fabricated from heats of material (i.e., both weld and plate) that are actually used in the beltline core region (Statement at ¶A.13). The coupon orientations are equivalent to the orientations of the specimens used to establish unirradiated impact properties (Id.). Three surveillance specimen capsules which contain both Charpy V-Notch and tensile specimens will be placed in the vessel (Id.). Each capsule also includes an Fe, Ni, and Cu flux wire (Id.). A separate neutron dosimeter is attached alongside one of the capsules and contains 3 Cu and 3 Fe wires (Id.). These dosimeters can be used to check periodically the actual flux field to which the capsules are exposed (Id.). Moreover, the specimens will be removed at intervals over the life of the vessel in order to confirm the adequacy of the predicted irradiation effects (Id.).

## Conclusion

In summary, PTS will not be a problem in the Limerick Generating Station due to the conservative design, selection of materials, construction, testing, operation, and surveillance of the Station, when combined with the physical behavior of a BWR. Applicant has met the requirements of 10 C.F.R. §2.749 by establishing that there is no genuine issue

as to any material fact and that it is entitled to a decision as a matter of law. The requested relief should be granted.

Respectfully submitted,

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October 12, 1983