UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

Before the Atomic Safety and Licensing Board

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

APPLICANT'S STATEMENT OF MATERIAL FACTS AS TO WHICH THERE IS NO GENUINE ISSUE TO BE HEARD AS TO CONTENTION I-62

1. 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).

2. Thermal stresses are produced in the vessel walls when cold water is introduced into the vessel (Id.).

3. Stresses are produced when the vessel is subjected to high pressures (Id.).

4. 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.).

5. The materials of which the reactor pressure vessel are made can become embrittled as a result of substantial neutron bombardment (Statement at ¶¶A.4 and A.9).

6. 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 (Statement ¶A.4).

7. Pressurized thermal shock 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.).

8. High reactor vessel pressure during cold water does not occur in a BWR (Statement at ¶A.5).

9. Significant neutron irradiation embrittlement does not occur in a BWR (Id.).

10. The decrease in vessel material fracture toughness as a result of irradiation is substantially less in a BWR than that in a PWR (Id.).

11. The pressure in a BWR follows the water-steam saturation curve (Id.).

12. During cold water injection, the pressure in a BWR drops because the water and steam remain in equilibrium (Id.).

13. 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.).

14. There is minimum radiation embrittlement of BWR vessel materials due to the factors stated in Statements 12 and 13 (Id.).

15. The Limerick reactor pressure vessels 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).

16. 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.)

17. Required weld test samples were performed for each procedure for major vessel full penetration welds (Id.).

18. 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.).

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

20. Fracture toughness is a material's inherent ability to resist unstable extension of flaws in the

presence of applied, dynamic, impact, or other suddenly applied loads (Statement at ¶A.7).

21. The fracture toughness properties of the vessel materials were measured and controlled in accordance with ASME Code Section III requirements to limits specified by the General Electric Company (GE) (Statement at ¶A.8).

22. 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).

23. Section 5.3.1.5 of the Final Safety Analysis Report demonstrates the compliance of the Limerick reactor vessel design with the requirements of 10 C.F.R. Part 50 Appendix G (Id; §5.3.1.5 FSAR).

24. 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.).

25. The ferrictic pressure boundary materials were also tested to the augmented fracture toughness requirements specified by GE (Id.).

26. All reactor pressure vessel components were impact tested by either the drop-weight test or the Charpy V-Notch impact test (Id.).

27. Both the drop-weight test and the Charpy V-Notch impact test were conducted on beltline plate material, closure flange material, top head material. feedwater and LPCI nozzle material forgings (Id.).

28. 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.

29. Compliance of the Limerick Unit 2 reactor with 10 C.F.R. Part 50 Appendix G will be demonstrated prior to issuance of an operating license for that unit (Id.).

30. The fracture toughness of a material can be affected by neutron fluence (Statement at ¶A.9).

31. The effect of neutron fluence over the life of the facility on the pressure vessel materials is a decrease in the fracture toughness of the materials (Id.).

32. The decrease in toughness is significant only at high fluences -- well above those expected in the Limerick vessels based upon conservative calculational techniques as verified by operational experience (Id.).

33. The neutron fluence calculations were made by using analytical models incorporating reactor core data,

geometric arrangement, and basic physical data (Statement at ¶A.10).

34. The reactor core data, geometric arrangement, and basic physical data were incorporated into computer programs to solve the transport equations for nuclear radiation (Id.).

35. The computer programs were verified in compliance with the quality assurance provisions of 10 C.F.R. Part 50 Appendix B (Id.).

36. The results of neutron fluence calculations have been compared to field measurements and have been found to overpredict conservatively the neutron fluence (Id.).

37. From the viewpoint of the neutron vessel fluence calculations, the BWR vessel geometry can be described in terms of a cross-sectional view of concentric cylinders (Id.).

38. The inner core region consists of 764 square fuel bundles of identical geometrical size arranged in a pattern simulating a cylinder (Id.).

39. The inner core region is surrounded by an annulus of water of an average thickness of approximately 8 inches (Id.).

40. The core water is separated from the downcommer water by a stainless steel metal shroud 2 inches thick (Id.).

41. The downcommer region containing the jet pumps is a water region 22.1 inches thick (Id.).

42. The reactor pressure vessel is approximately 6.2 inches thick in the beltline region (Id; FSAR Figure 4.3-29).

43. A maximum fluence for the core beltline material of 1.1×10^{18} n/cm^2 at one quarter of the vessel wall thickness from the inside diameter was conservatively calculated (Statement at ¶A.11).

44. This maximum beltline 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.).

45. For critical nozzles, such as the LPCI nozzle, the fluence was uniquely calculated for each nozzle (Id.).

46. The resulting core beltline operating limitations are less restrictive than those operating limitations established as a result of analyses of other reactor parts (Id; FSAR Figure 5.3-4).

47. 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.).

48. FSAR Figure 5.3-4, which shows the results of the above calculations, will become part of the technical specifications for Limerick Unit 1 and will define the minimum temperature versus reactor pressure required to assure adequate fracture toughness (Id.).

49. A fracture mechanics evaluation of the effects of a potential loss of coolant accident in a BWR/6 reactor vessel of design similar to Limerick was performed (Statement at ¶A.12).

50. The fracture mechanics evaluation determined the thermal stresses in the vessel and calculated the resulting applied stress intensity factors which would exist at the tip of a potential crack in the vessel wall (Id.).

51. 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.).

52. Available vessel material fracture toughness always exceeded the applied stress intensity factors for all postulated crack depths (Id.).

53. Crack propogation will not occur even for large initial flaws with depths approaching the vessel wall thickness (Id.).

54. Limerick specific fluence, material properties, and NDT shift data confirm the results of the fracture mechanics evaluation as applicable to the Limerick reactor pressure vessels (Id.).

55. 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).

56. The surveillance coupon orientations are equivalent to the orientations of the specimens used to establish unirradiated impact properties (Id.).

57. Three surveillance specimen capsules which contain both Charpy V-Notch and tensile specimens will be placed in the vessel (Id.).

58. Each surveillance specimen capsule also includes an Fe, Ni, and Cu flux wire (Id.).

59. A separate neutron dosimeter is attached alongside one of the capsules and contains three Cu and three Fe wires (Id.).

60. The neutron dosimeters can be used to check periodically the actual flux field to which the capsules are exposed (Id.).

61. The surveillance specimens will be removed at intervals over the life of the vessel in order to confirm the adequacy of the predicted irradiation effects (Id.).

Respectfully submitted,

CONNER & WETTERHAHN, P.C.

Mark J. Wetterhahn

Mark J. Wetterhahn
Counsel for Philadelphia
Electric Company

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