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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

Before the Atomic Safety and Licensing Board

In the Matter of	)	
	)	Docket No. 50-352-OLA
Philadelphia Electric Company	)	(Check Valve)
	)	Docket No. 50-352-OLA-2
(Limerick Generating Station,	)	(Containment Isolation)
Unit 1)	)	March 26, 1986

LICENSEE'S ANSWER TO CONTENTION SUPPLEMENTS PROPOSED BY INTERVENOR ROBERT L. ANTHONY ON AMENDMENT NO. 1 AND CONTENTIONS PROPOSED ON AMENDMENT NO. 2

Preliminary Statement

This matter concerns the late-filed petition of Robert L. Anthony in response to the notice of opportunity to request a hearing on a proposed amendment of the operating license for the Limerick Generating Station, Unit 1 ("Limerick").1/ The NRC Staff issued the requested amendment as Amendment No. 2 to the Limerick operating license on March 3, 1986.

Although the notice in the Federal Register stated that a timely request for intervention must be filed by January 29, 1986,2/ Mr. Anthony did not file his petition until February 26, 1986. The Licensee opposed Mr. Anthony's

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1/ 50 Fed. Reg. 52874 (December 26, 1985).

2/ The notice originally misstated the deadline as February 3, 1986 but was corrected by a subsequent notice. See 51 Fed. Reg. 1051 (January 9, 1986).

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late-filed petition regarding Amendment No. 2 for failure to satisfy or even discuss the lateness criteria and for lack of standing.[3/]

On March 14, 1986, the presiding Atomic Safety and Licensing Board entered an order consolidating the proceedings on Amendment No. 1 and Amendment No. 2 and scheduled a prehearing conference for March 27, 1986.[4/] The Board directed Mr. Anthony "to file his containment-isolation contentions and any amended petitions or supplements to his containment-isolation petition by express mail no later than March 20, 1986."[5/] The Board stated that the NRC Staff and the Licensee would be afforded an opportunity at the prehearing conference to address Mr. Anthony's contentions and any amended petition or supplement to his petition.[6/]

3/ Because the NRC Staff had not been served with Mr. Anthony's second petition and received an extension of time to answer from the Secretary, the Board afforded the Staff an opportunity to answer the petition orally at the prehearing conference on March 27, 1986. See Philadelphia Electric Company (Limerick Generating Station, Unit 1) (Check Valves) (Containment Isolation), "Memorandum and Order Consolidating Proceedings and Setting Schedule for Identification of Issues" (March 14, 1986) (slip op. at 5).

4/ Id. at 4.

5/ Id. at 5.

6/ Id. We assume that the Board did not intend to preclude Licensee or the Staff from filing a written response on an expedited basis if either chose to do so.

For the reasons discussed below, Licensee opposes the contentions and contention supplements proposed by Mr. Anthony on the grounds that they lack the requisite specificity and bases, fail to state any litigable issue and exceed the scope of this proceeding.

Argument

In answer to the contentions proposed by Mr. Anthony in the proceeding on Amendment No. 1, Licensee made initial observations of general applicability regarding the admissibility of contentions. Specifically, Licensee noted that the scope of the proceeding is narrowly limited to questions relating to the validity of Amendment No. 1 and could not be a basis for litigating safety and environmental issues which were or could have been litigated in the operating license proceeding for Limerick; that the petitioner had had ample advance notice to prepare contentions, especially considering the brevity of the record; and that the proposed contentions were so completely lacking in technical specificity that it was impossible to understand their substance or determine the regulation or other requirement with which Licensee allegedly had failed to comply.

As discussed below, the same points apply with equal force to Mr. Anthony's contention supplement regarding Amendment No. 1 and his proposed contentions on Amendment No. 2. Licensee therefore respectfully refers the Board to its earlier answer as if fully incorporated herein.[7/] With these basic principles as background, Licensee now addresses each of Mr. Anthony's proposed contentions.

Supplementary Contentions Relating to Amendment No. 1

Contention 1 (Supp.). This is another example of Mr. Anthony's *ipse dixit* that the NRC should not have made the "no significant hazards consideration" finding for Amendment No. 1 because of some unspecified "increased risk of plant failure and radioactive releases"[8/] which is nowhere explained. Because this allegation completely lacks specificity and basis, it adds nothing to his original argument that an environmental impact statement or environmental assessment was necessary for issuing the amendment.

Contention 3 (Supp.). Mr. Anthony alleges that the extension granted by Amendment No. 1 (which he erroneously cites as eight months)[9/] will somehow result in "wear and

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7/ "Licensee's Answer to Contentions Proposed by Intervenor Robert L. Anthony" at 2-10 (March 17, 1986). Licensee also concurs in the objections raised by the NRC Staff in opposing Mr. Anthony's contentions regarding Amendment No. 1, which it filed on March 17, 1986, and which are likewise applicable here.

8/ Supplement to Anthony Contentions at 1 (March 19, 1986).

9/ *See* 50 Fed. Reg. 52874 (December 26, 1985) (extension will be "a maximum of 96 days beyond the time otherwise designated" by plant Technical Specifications). Technical Specification 4.0.2 (copy attached) permits a 25 percent extension of each surveillance interval. Thus, the 96-day extension is correct.

aging of the valves and instrument lines."[10/] No basis is cited for this allegation. Mr. Anthony has stated nothing to challenge the validity of the explanation in FSAR §6.2.4.3.1.5 (copy attached), as restated in Licensee's Answer to his initially filed contentions, that the "excess flow check valves in question are passive and do not operate unless a breach occurs in the instrumentation line such that the resulting flow causes a differential pressure across the valve."[11/] Accordingly, the contention is still deficient.

Contention 5. Mr. Anthony alleges that the check valves could "stick open, stick closed, rupture or separate from the pipe,"[12/] but fails to relate this generalization to the design of the excess flow check valves or the requested extension of time for leakage testing. As in Contention 3 (Supp.), he provides no basis for the assertion that these valves will "stick open" during the 96-day extension. Were an excess flow check valve to "stick closed," this fact would be indicated in the control room and, in any event, would be in a safe direction with regard to leakage. Moreover, an extension of time for leakage

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10/ Supplement to Anthony Contentions at 1 (March 19, 1986).

11/ "Licensee's Answer to Contentions Proposed by Intervenor Robert L. Anthony" at 14 (March 17, 1986).

12/ Supplement to Anthony Contentions at 1 (March 19, 1986).

testing is unrelated to "rupture or separa[tion] from the pipe."[13/] While Mr. Anthony alleges that "[e]xcessive coolant pressure"[14/] could cause such failure mechanisms, he nowhere specifies any mechanism for this or relates it to excess flow check valve testing. As has been previously discussed, the consequences of the failure of an excess flow check valve to operate in conjunction with a hypothesized break of the lines in the reactor enclosure was analyzed during the operating license stage of this proceeding and the consequences found to be well within the guideline values of 10 C.F.R. Part 100.

The inspection report cited by Mr. Anthony does not provide a basis for Contention 5. Inspection Report No. 50-352/86-02 (copy attached) at pages 4-5 discusses condensation of steam in the turbine building, a non-safety related structure. Mr. Anthony provides no nexus between the reported incident and the hypothesized accidents in the reactor enclosure, a safety related structure which forms the secondary containment boundary. This Inspection Report at 9 also discusses a minor overflow from a sample sink located outside the secondary containment into a floor drain. Mr. Anthony fails to develop any relationship between this occurrence and any deficiency in the analysis

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13/ Id.

14/ Id.

of the failure of an excess flow check valve contained in the Final Safety Analysis Report.

Mr. Anthony again attempts to raise the question of interfacing LOCAs as the result of a failure of an excess flow check valve to close.[15/] What Mr. Anthony fails to appreciate is that the instrument lines downstream of the check valves are not designed for lower pressures and temperatures than on the upstream side.[16/] In fact, because excess flow check valves are normally open, instrument lines are designed for and exposed to primary system pressures during operation. Mr. Anthony attempts to utilize data on other valves provided by the Applicant in support of Amendment No. 2 to support his assertion as to the lack of reliability of the excess flow check valves. However, the data relates to entirely different types of valves and manufacturers.[17/] Mr. Anthony has provided no reason why such data is applicable to excess flow check valves. The contention, even as supplemented, is inadmissible.

15/ Id.

16/ It is not at all clear that Mr. Anthony is utilizing the term "interfacing LOCA" as generally utilized in the industry. In this contention, he apparently is trying to raise as a contention the effect of an instrumentation line rupture on surrounding systems in the reactor enclosure. This matter, which was considered at the operating license stage, is clearly beyond the scope of the Amendment.

17/ The excess flow check valves for Limerick Unit 1 are
(Footnote Continued)

Contention 11 (Supp.). In this supplement, Mr. Anthony simply repeats the same misassumption stated in the original contention, i.e., that Licensee could have conducted the required testing during periods when the reactor was shut down in 1985 or early 1986. As Licensee has demonstrated, however, the tests require about two weeks to perform,[18/] and the mere summing of reactor down days does not show that a two-week period was available. Therefore, the contention, even as supplemented, still lacks any basis. In any event, it is irrelevant whether testing could have been performed at any other time. The only question is whether the amendment as granted meets applicable safety and environmental regulations.

Contention 12. In this new contention, Mr. Anthony asserts that reactor instrumentation and controls necessary for shutting down the reactor might be unavailable in the event of "an instrument line or valve rupture."[19/] Although Mr. Anthony cites a portion of the Staff's Safety Evaluation which states that the instrument line may "serve [as] an

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(Footnote Continued)
manufactured by the Marotta Company. Thus, data on different valves can provide no basis or specificity regarding this contention.

18/ "Licensee's Answer to Contentions Proposed by Intervenor Robert L. Anthony" at 22 (March 17, 1986).

19/ Supplement to Anthony Contentions at 2 (March 19, 1986).

instrumentation manifold with multiple transmitters,"[20] nowhere does Mr. Anthony cite any basis for assuming that the failure of an instrument line could jeopardize the capacity for safe shutdown. Indeed, Licensee's Response to Question 421.10 specifically analyzes why an instrument line break would not interfere with reactor scram and safe shutdown.[21] In any event, a reexamination of this issue is irrelevant to the extension granted for testing excess flow check valves and is therefore clearly beyond the scope of the proceeding. This contention should be denied.

Contention 13. In this contention, Mr. Anthony asserts that PECO's application for Amendment No. 1 and the Staff's Safety Evaluation contain unsupported assumptions as to the reliability of the subject excess flow check valves in light of industry experience to date. This contention is without basis on its face. PECO's application, dated December 18, 1985, states at pages 5 and 6 as follows:

> A review of the Nuclear Plant Reliability Data System and a poll of several utilities having similar make and model valves revealed no instances of the valves failing to perform their safety-related function. During the first surveillance test, all valves tested successfully. Philadelphia

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20/ Id., citing Amendment No. 1 Safety Evaluation at 2.

21/ Question 421.10 (copy attached) speaks to "common Taps, hydraulic headers and impulse lines feeding pressure, temperature, level or other signals to two or more control systems." See also Response to Question 421.11 (copy attached).

> Electric's Peach Bottom Units 2 and 3 have valves which are similar in design, although by a different manufacturer, and have had a high degree of success with these valves checking properly.

Similarly, the Staff stated at page 2 of its Safety Evaluation:

> The licensee has examined the records of the initial flow testing performed on these valves and found that all valves were tested successfully. The licensee further states that, based on available data, the valves are believed to be highly reliable in performing their function of checking flow. The staff concludes that the condition of the valves is not expected to change significantly during the short extension period.

Contentions Relating to Amendment No. 2

Contention 14. This proposed contention simply states Mr. Anthony's "disagreement with findings by NRC."[22/] No admissible issue is stated simply by an intervenor's noting his "disagreement" with the NRC's Safety Evaluation.

Contention 15. Similar to Mr. Anthony's argument in Contentions 3 and 10 that no amendment from existing license conditions could be granted, this contention asserts that "no basis in the regulations" authorizes the Staff to find that an exemption is warranted.[23/] As stated at page 6 of the Exemption, the Staff acted pursuant to 10 C.F.R. §50.12.

22/ Supplement to Anthony Contentions at 3 (March 19, 1986).

23/ Id. at 3-4.

Mr. Anthony's contention that the extension "is not warranted"[24/] totally lacks any basis or specificity and fails to state a litigable issue.

Contention 16. Mr. Anthony's assertion that nothing in the regulations prevents a shutdown for tests fails to state any reason why it would be necessary for the Staff to require an unnecessary and avoidable shutdown by declining to grant the requested extension. No admissible issue has been stated here. The remainder of this proposed contention regarding the record of valve performance ignores the Staff's findings at pages 3 and 4 of the Exemption, which states:

> The staff reviewed available data provided by the licensee on similar valves used elsewhere in the industry which supports the licensee's position that these valves have traditionally good maintenance histories in the industry. The staff also reviewed previous leakage test results on the specific valves subject to the exemption request and has found that there is substantial margin between the values previously measured and the limiting values in Appendix J and the Technical Specifications to accommodate any additional degradation likely to occur during the period of the extension.
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> . . .
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> The Licensee has provided various bases for its conclusion that the requested delay of 12 weeks is not likely to result in a situation wherein the measured leakage from these valves would cause the limitations of the technical

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24/ Id. at 4.

> specifications to be exceeded. These bases, which are discussed in more detail in the enclosed Safety Evaluation and the licensee's submittals, include the licensee's characterization of these valves as being of the type which traditionally have good maintenance histories, are not used in the relatively more demanding applications and which have shown in their initial leakage tests that they do not contribute an undue proportion of either the total measured containment leakage or the technical specification allowable leakage values.

Mr. Anthony provides no allegations with requisite basis and specificity to contest these conclusions.

Contention 17. Mr. Anthony takes exception to findings contained in the Staff evaluation of the exemption associated with Amendment No. 2. He asks that the Board reject outright the conclusions reached by the Staff but does not provide a basis for doing so. Licensee Event Report ("LER") 85-102 (copy attached), which Mr. Anthony cites in passing, does not provide such a basis. Mr. Anthony does not state why an isolated incident concerning a single valve provides sufficient basis to call into question the leak tightness of the valves which are the subject of the amendment request. This contention lacks basis and should be denied.

Contention 18. In this contention, Mr. Anthony continues to attempt to draw conclusions from the facts reported by the Licensee in LER 85-102. The problem with his argument is that merely because certain letters or numbers in valve designations utilized at Limerick are similar, as cited by Mr. Anthony, does not mean that the valves are of the same manufacturer, type, size or design. Thus, it is not proper to draw conclusions simply based upon the valve designation number. In any event, the design criteria for Limerick as required by 10 C.F.R. Part 50 anticipates single failures of components such as valves. As noted in the LER, the line in question had both inboard and outboard isolation valves. Merely because a single valve leaked would not affect the ability to safely shut down the reactor or mean that offsite consequences would be increased. This contention should be denied for lack of basis.

Contention 19. At the outset, Mr. Anthony again misstates the actual length of the extension granted for Amendment No. 2.[25/] The basic assumption of this contention is that the extension of the test interval causes "probabilities for faults in the valves [to] accelerate."[26/] This assertion is without basis. Mr. Anthony's manipulation of the raw data provided by Licensee lacks any basis whatsoever.

Mr. Anthony alleges that there have been six failures out of a total of 61 valves manufactured by Atwood and Morrill. From this he concludes that "there is a 50%

25/ See 50 Fed. Reg. 53235 (December 30, 1985) (extension will permit surveillances to be performed "a maximum of 84 days" beyond the time otherwise required).

26/ Supplement to Anthony Contentions at 5 (March 19, 1986).

possibility that one of the 5 valves above could fail in the period of extention [sic] to the testing schedule."[27/] However, meaningful statistics cannot be drawn until the period for which the 61 valves operated and in which these six failures occurred is known. Moreover, the note to the table from which Mr. Anthony drew this information clearly states that the reason for failures was attributable to "wear out," a failure mechanism which would not be applicable to a new plant such as Limerick. Moreover, as noted in the table, the data did not relate to the specific model, size and type of valve installed at Limerick.[28/] Thus, there is no basis for the admission of this contention.

Contention 20. This contention asserts the same theory shown to be invalid for Contention 11. Mr. Anthony asserts that there were 40 days from August to January when the plant was shutdown and tests could have been performed. As stated in the application and the Staff's Safety Evaluation, it would be necessary to shut the plant down for a two-week period in order to perform the necessary tests for which an

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27/ Id.

28/ See letter to Mr. Robert Bernero, Director, Division of Boiling Water Reactor Licensing, from M.J. Cooney, Manager, Nuclear Production, Philadelphia Electric Company, dated January 29, 1986 (copy attached).

extension was granted in Amendment No. 2.[29/] Mr. Anthony has asserted no basis for challenging those statements.

Contention 21. In this contention, Mr. Anthony simply asserts error in the Staff's application of 10 C.F.R. §50.12. An intervenor who simply asserts "the opposite"[30/] of what the Staff has determined in its licensing action has wholly failed to state any litigable issue.

Contention 22. Likewise, this assertion that the exemption and amendment will have a significant impact on the environment merely states Mr. Anthony's conclusion and provides no supporting basis or specificity. For the reasons stated previously with regard to proposed Contentions 1 and 2, this contention states no basis for litigation.

Contention 23. This contention simply takes issue with certain correspondence which in part forms the basis for the Staff's action. The fact that Mr. Anthony would call these matters into question, however, is no basis for litigating whether or not the exemption and amendment should have been

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29/ See Application for Amendment of Facility Operating License NPF-39 and Exemption to Part 50, Appendix J at 2 (December 18, 1985); Safety Evaluation by the Office of Nuclear Reactor Regulation, Support Amendment No. 2 to Facility Operating License No. NPF-39 at 2 (March 3, 1986); Limerick, supra, "Exemption" at 2 (March 3, 1986).

30/ Supplement to Anthony Contentions at 5 (March 19, 1986).

issued. An intervenor cannot shift his burden of meeting the requirements for specificity and basis under Section 2.714(b) by simply claiming that licensing actions are unwarranted or unsubstantiated.

Contention 24. Again, as with Contention 14, Mr. Anthony merely states his disagreement with the Staff's conclusion and thus presents no admissible issue. His assertion that "the figures in the NPRDS tables . . . do not support the conclusion that these valves should not [experience undue difficulties in meeting the leakage criteria]"31/ is likewise without basis and specificity.32/

Contention 25. In this contention, Mr. Anthony repeats assertions previously made in Contentions 8 and 19, citing a study by Sarah M. Davis in support of his assertion that residual heat removal and low pressure coolant injection lines and valves are vulnerable to an interfacing LOCA. He provides no basis or further specificity whatsoever to

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31/ Id. at 6.

32/ The reference, an NRC press release, also adds no basis for the contention. There has been no showing that the valves discussed in the press release are of the same type or in the same service as the valves at Limerick covered by Amendment No. 2. In fact, no feedwater check valves are involved; check valves covered by the Amendment are "normally closed."

support his "exact opposite conclusion"[33/] from that reached by the Staff in its Safety Evaluation.

Initially, the technical specification for which an extension has been sought does not relate to interfacing LOCAs for Limerick. That subject is addressed in a separate technical specification for which no extension or exemption is required, namely, Section 3.4.3.2 (copy attached). The question of high pressure - low pressure systems interface was addressed at the operating license stage in the Final Safety Analysis Report.[34/] The operator has available to him various indications of such valve leakage from high pressure to low pressure systems. Thus, this matter is not within the scope of the requested amendment or exemption and should therefore be excluded as a contention.

Contention 26. This proposed contention, too, merely states Mr. Anthony's disagreement with the Staff's findings. Clearly, no admissible issue is stated.

Contention 27. Here, Mr. Anthony asks the Board to find that Amendment No. 2 involves a significant hazards consideration, challenges the Staff's environmental finding of no significant impact and states that the environmental findings contained in the Amendment do not meet the

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33/ Supplement to Anthony Contentions at 6 (March 19, 1986).

34/ *See* FSAR Section 7.6.1.2 and Response to Question 421.50 (copies attached).

requirements of the NRC's regulations. Absolutely no basis or specificity is provided in support of this allegation and it is inadmissible.

Contention 28. Mr. Anthony baldly asserts "the opposite conclusion to the staff's"[35/] that the health and safety will not be endangered by the grant of the Amendment and that the activities under the Amendment will be conducted in compliance with the Commission's regulations. No basis or specificity is provided and this contention states no basis for litigation.

Contention 29. Mr. Anthony states that he has been provided no evidence to indicate that the Commission has met its obligation of State consultation, as provided in 10 C.F.R. §50.91(b) and requests that the Board thus find that the Amendments were issued in violation of the regulations. Mr. Anthony has not shown that the Commission has failed to meet its responsibilities in this regard. As noted with regard to Contention 23, supra, intervenor cannot shift his burden of meeting the requirements for specificity and basis under Section 2.714(b) by simply claiming that licensing actions are unsubstantiated. In any event, such a contention is beyond the scope of his interest in this proceeding and should be denied.

35/ Supplement to Anthony Contentions at 7 (March 19, 1986).

Conclusion

For the reasons discussed above, each of the contentions proposed by Mr. Anthony lacks the requisite specificity and bases under 10 C.F.R. §2.714(b) and should not be admitted. Further, a number of the proposed contentions exceed the limited scope of this proceeding or impermissibly challenge NRC regulations. Inasmuch as Mr. Anthony has failed to plead a single admissible contention, his petition should be dismissed.[36]

Respectfully submitted,

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March 26, 1986

36/ See 10 C.F.R. §2.714(b); Duquesne Light Company (Beaver Valley Power Station, Unit 2), LBP-84-6, 19 NRC 393, 395, 430 (1984).

NUREG-1149

Technical Specifications
Limerick Generating Station, Unit No. 1

Docket No. 50-352

Appendix "A" to
License No. NPF-39

**Issued by the
U.S. Nuclear Regulatory
Commission**

Office of Nuclear Reactor Regulation

June 1985

APPLICABILITY

SURVEILLANCE REQUIREMENTS

4.0.1 Surveillance Requirements shall be met during the OPERATIONAL CONDITIONS or other conditions specified for individual Limiting Conditions for Operation unless otherwise stated in an individual Surveillance Requirements.

4.0.2 Each Surveillance Requirement shall be performed within the specified time interval with:

a. A maximum allowable extension not to exceed 25% of the surveillance interval, but

b. The combined time interval for any 3 consecutive surveillance intervals shall not exceed 3.25 times the specified surveillance interval.

4.0.3 Failure to perform a Surveillance Requirement within the specified time interval shall constitute a failure to meet the OPERABILITY requirements for a Limiting Condition for Operation. Exceptions to these requirements are stated in the individual Specificatons. Surveillance requirements do not have to be performed on inoperable equipment.

4.0.4 Entry into an OPERATIONAL CONDITION or other specified applicable condition shall not be made unless the Surveillance Requirement(s) associated with the Limiting Condition for Operation have been performed within the applicable surveillance interval or as otherwise specified.

4.0.5 Surveillance Requirements for inservice inspection and testing of ASME Code Class 1, 2, & 3 components shall be applicable as follows:

a. Inservice inspection of ASME Code Class 1, 2, and 3 components and inservice testing of ASME Code Class 1, 2, and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR Part 50, Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 50, Section 50.55a(g) (6) (i).

b. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice inspection and testing activities required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these Technical Specifications:

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice inspection and testing activities	Required frequencies for performing inservice inspection and testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days

6.2.4.3.1.5 Evaluation Against Regulatory Guide 1.11

Instrument lines that penetrate the containment from the RCPB conform to Regulatory Guide 1.11 in that they are equipped with a restricting orifice located inside the drywell and an excess flow check valve located outside and as close as practicable to the containment. Should an instrument line that forms part of the reactor pressure boundary develop a leak outside the containment, a flow rate that results in a differential pressure across the valve of 3 to 10 psi causes the excess flow check valve to close automatically. Should an excess flow check valve fail to close when required, the main flow path through the valve has a resistance to flow at least equivalent to a sharp-edged orifice of 0.375 inch diameter. Valve position indication is provided in the reactor enclosure. Those instrument lines that do not connect to the RCPB conform to Regulatory Guide 1.11 in that they are either equipped with an excess flow check valve or an isolation valve capable of remote operation from the control room, and are sized or orificed to meet the criteria outlined in Regulatory Guide 1.11. The drywell pressure, suppression pool level, suppression chamber pressure, and drywell sump level instrument lines are:

a. Provided with isolation valves capable of remote operation from the control room.

b. Q-listed, as discussed in Section 3.2.

c. Designed to seismic Category I standards.

d. Designed to withstand containment design pressure and temperature.

e. Terminate in the reactor enclosure, which is served by the SGTS.

The status of the isolation valves capable of remote operation from the control room is indicated in the control room.

The TIP system lines as shown in Figure 9.3-2 and described below are considered instrument lines because (a) they function as instrument lines or support the operation of instrument lines, and (b) they are small diameter lines.

TIP system isolation valves are provided on each guide tube immediately outside the containment. Dual barrier protection is provided by a solenoid operated ball valve and an explosive actuated cable shearing valve. The ball valve is closed except when a TIP is inserted. These valves prevent loss of reactor coolant in the event that an incore guide tube ruptures inside the reactor vessel and prevents the escape of primary containment atmosphere.

The guide tube ball valve solenoid is normally de-energized and the valve is in the closed position. When the TIP starts forward, the valve solenoid is energized and the valve is held open against its spring. As the valve opens, it actuates a set of contacts which provide position indication at the TIP control panel and a permissive signal for TIP motion. Upon receipt of a containment isolation signal (reactor low water level or high drywell pressure), the TIP drive mechanism is signalled to retract the TIP. As the TIP is withdrawn into its shield chamber outside containment, a position switch signals the ball valve to close.

The shear valve is provided as a backup in the event that a TIP cannot be retracted or a ball valve sticks open when containment isolation is required. In this event, the shear valve would be operated from the control room to cut the cable and seal the guide tube. Continuity of the shear valve squib firing circuits is continuously monitored by front panel indicator lights in the control room.

The guidelines of Regulatory Guide 1.11, Section 1.b are met for the TIP system as discussed below.

An analysis of the maximum leakage rate from the TIP system and the offsite radiological effects under normal reactor operating conditions was performed. The analysis conservatively assumed that all TIP system lines suffered guillotine breaks just outside the containment boundary. Specific activity inside the primary containment was assumed to be at the maximum technical specification limit for iodine in the primary coolant. (This is an extreme conservatism because a primary coolant rather than drywell atmosphere source term was assumed.) To characterize maximum flow through the TIP system lines, the drywell was assumed to be at its maximum normal pressure (2.0 psig) and normal temperature (135°F). It was also conservatively assumed that all TIP probes are fully retracted. Under these conditions, total flow from the TIP system lines would be only 0.105 lbm/sec as compared to 2.2 lbm/sec for an instrument line which penetrates the reactor primary coolant boundary. The corresponding 24-hour site boundary dose for this flow rate (using worst case average annual meteorology) would be less than 0.03 rem thyroid. The conservatively calculated leak rate is extremely low and the offsite dose is a small fraction of 10CFR100 limits.

The TIP guide tubes are equipped with dual isolation valves located as close to containment as practical; a solenoid actuated ball valve and an explosively actuated shear valve acting in series. The ball valves are normally de-energized (in a closed position). Consequently, during normal operation, the containment isolation function for the TIP system is accomplished without the need for any action. Therefore, requirement of Regulatory Guide 1.11 Section 1.C.1 is met. In the unlikely event of a LOCA while the TIP system is in operation, containment isolation is automatically accomplished as follows. Upon receipt of a containment isolation signal (reactor low water level or high drywell pressure), the TIP drive mechanism is automatically signaled to retract the TIP. As the TIP is withdrawn into its shield chamber, a position switch signals the ball valve to close. All TIP line ball valves open against a spring and will close on loss of power. The cable shearing valves are equipped with redundant explosive actuating devices increasing the isolation reliability of the system and are remote manually operated from the control room. The ball and shear valves are instrumented to indicate position (i.e., open or closed).

Accidental closure of the TIP line isolation valves does not create a safety hazard, nor is the TIP system required to operate during an accident to mitigate the consequences of that accident. Therefore, the isolation provisions of the TIP system comply with the requirements of Regulatory Guide 1.11 Section 1.C.2. When the TIP starts forward, the ball valve solenoid is energized and the valve is held open against its spring. This satisfies the requirement of Regulatory Guide 1.C.3, and therefore satisfies all the requirements of Regulatory Guide 1.11 Section 1.C.

The design of the TIP isolation system is commensurate with the importance to safety of isolating that system. It recognizes that the TIP system design is such that the TIP guide tube isolation ball valve is normally closed. Typically, a TIP scan requires insertion of the TIP probes into the reactor vessel for a period of approximately four hours per month. Over a one-year period, this amounts to a total of 120 hours per year, or less than 2% of the time.

Because of the normally closed state of the TIP ball valves, the probability of a release of radioactivity through the TIP guide tubes following a LOCA is extremely low. Even in the event of a LOCA, the TIP system design will reliably provide automatic isolation of any open TIP guide tubes by providing for automatic retraction of the TIP cable followed by automatic closure of the TIP ball valve. Should the ball valve fail to automatically close, that condition would be indicated to the operator in the control room. The operator could then manually actuate the shear valve in the control room to isolate that line.

The design of the TIP system isolation provisions is based on the low probability that the system will be called upon to isolate the containment following a significant fission product release to the containment atmosphere. Consequently, the power supplies and the controls for the TIP isolation valves are not safety grade. However, the overall system reliability for isolation is high because: (1) the ball and shear valves are powered from separate power supplies, (2) the shear valves are powered from an onsite dc power source, (3) the ball, shear, and purge check valves, and the line from the containment to the outermost isolation valve are mechanically safety grade, (4) upon loss of power the ball valves close, and (5) the TIP system receives automatic LOCA signals to retract and isolate.

In considering the potential magnitude of a fission product release through the TIP guide tubes as a result of a design basis LOCA event, it is appropriate to consider the event probability. There are several sequences of events which could lead to a fission product release through the TIP guide tubes for a degraded core event. These are shown in the event tree diagram on Figure 6.2-53. Any sequence which leads to such a release must involve at least two events: (1) a loss-of-coolant accident, and (2) a degraded core. The probability of this combination of events alone is on the order of 10^{-7} per reactor year. Additional failure(s) in the TIP isolation system have to be assumed in order to have a radiological release through the TIP lines. These failures and their associated probabilities are shown on the event tree diagram, Figure 6.2-53.

As shown on the event tree diagram, the most likely sequence leading to fussion product release through the TIP guide tubes is Event N. The probability of occurrence of this event/failure sequence is about 5×10^{-13} per reactor year. This analysis assumes the proper functioning of non-safety grade power supplies and circuits for the TIP isolation valves in determining overall system reliability. The low probability of a fission product release to the environment through the TIP guide tubes demonstrates the adequacy of the current TIP isolation system design basis.

Although the above discussion indicates an extremely low likelihood of a fission product release through the TIP guide tubes or purge lines, the consequence of that release has been evaluated for the most probable event. That event would involve the failure of all five TIP guide tubes to isolate following a degraded core event. In this instance, the TIP probe substantially reduces the flow area in the TIP guide tube which provides a pathway for fission product release unless some other unlikely event (i.e., earthquake) were to occur at the same time and cause further equipment failures. The pathway for an atmospheric fission product release would be through the check valve in the indexer box (open due to a positive internal containment pressure), down the long and narrow annulus between the TIP probe/cable assembly and the guide tube, and out through the end of the guide tube located in the reactor enclosure. The probe/cable assembly are never completely withdrawn from the guide tube, so the annular flow restriction is maintained. For the radiological analysis, Regulatory Guide 1.3 source terms and accident meteorology were assumed. The use of Regulatory Guide 1.3 source terms available for release is extremely conservative because it neglects any fission product plateout or fallout in the containment or in the constricted TIP tubes.

The results of the radiological evaluation show that the site boundary and low population zone doses for this limiting event using Regulatory Guide 1.3 assumptions are below 10CFR100 limits.

The low probability of fission product release and the results of the radiological evaluation satisfy the intent of Regulatory Guide 1.11 Section 1.d.

6.2.4.3.1.6 Evaluation Against Regulatory Guide 1.141

The containment isolation system conforms to Regulatory Guide 1.141 except as discussed below:

<u>Section 3.6.4</u> Single Valve and Closed System Both Outside Containment...

The single valve and piping between the containment and the valve shall be enclosed in a protective leaktight or controlled leakage housing to prevent leakage to the atmosphere.

<u>Limerick Design</u>:

For systems that fall into this category except for the ECCS pump suction lines, the single valve outside primary containment is not enclosed in a protective leaktight or controlled leakage housing. Moderate energy lines that fall into this category do not connect to the reactor coolant pressure boundary and are not postulated to break concurrent with a LOCA. Therefore, neither reactor coolant nor post-LOCA containment atmosphere are released. However, any leakage is contained within the secondary containment and is diluted and filtered prior to release. The ECCS pump suction isolation valves are enclosed in pump rooms adjacent to the containment that have provisions for the environmental control of any fluid leakage.

<u>Section 3.6.5</u> Two Valves Outside Containment...

The valve nearest the containment wall and piping between the containment and that valve shall be enclosed in a protective

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
631 PARK AVENUE
KING OF PRUSSIA, PENNSYLVANIA 19406

FEB 06 1986

Docket No. 50-352

Philadelphia Electric Company
ATTN: Mr. S. L. Daltroff
Vice President, Electric Production
2301 Market Street
Philadelphia, Pennsylvania 19101

Gentlemen:

Subject: Inspection No. 50-352/86-02

A routine safety inspection was conducted on January 6-10, 1986 of the Limerick Unit 1 radiation protection program. Areas that were reviewed included organization and staffing, actions related to a steam leak in the condenser bay, routine surveys, HP technician training, radiation work permits, and the start-up test program.

No violations were identified. No reply to this letter is required.

Your cooperation with us is appreciated.

Sincerely,

Ronald R. Bellamy for
Thomas T. Martin, Director
Division of Radiation Safety
and Safeguards

Enclosure: NRC Region I Inspection Report Number 50-352/86-02

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U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 50-352/86-02

Docket No. 50-352

License No. NPF-39 Priority -- Category C

Licensee: Philadelphia Electric Company
2301 Market Street
Philadelphia, PA 19101

Facility Name: Limerick Generating Station, Unit 1

Inspection At: Limerick, PA

Inspection Conducted: January 6-10, 1986

Inspectors: T. Dragoun, Radiation Specialist — Feb 4 1986 date

M. Miller, Radiation Specialist — 2/4/86 date

J. Kottan, Senior Radiation Specialist — 2/4/86 date

Approved by: M. Shanbaky, Chief, Facilities Radiation Protection Section — 2/4/86 date

Inspection Summary: Inspection on January 6-10, 1986 (Report No. 50-352/86-02)

Areas Inspected: Routine unannounced safety inspection of the licensee's radiation protection program including: organization and staffing; evaluation of a gaseous effluent release; actions taken for a steam leak in the condenser bay; routine radiological surveys; health physics technician training; radiation work permits; and the start-up test program. The inspection involved 68 inspector hours onsite by three region-based inspectors.

Results: No violations were identified.

DETAILS

1.0 Persons Contacted

1.1 Licensee Personnel

G. Leitch, Plant Manager
J. Spencer, Superintendent - Plant Services
R. Dubiel, Senior Health Physicist
J. Wiley, Senior Chemist
C. Endiss, Regulatory Engineer
J. Fongheiser, Radiation Protection Physicist
C. Harmon, Quality Assurance Engineer
G. Murphy, Technical Support HP
J. Rubert, Site Supervisor, EPQA
R. Titolo, Applied Health Physicist
V. Warren, Test Engineer

1.2 NRC Personnel

E. Kelley, Resident Inspector

All personnel listed above attended the exit interview on January 10, 1986.

Other licensee or contractor employees were also contacted or interviewed during this inspection.

2.0 Purpose

The purpose of this routine inspection was to review the licensee's radiation protection program with respect to the following elements:

- Status of previously identified items
- Organization and staffing
- Evaluation of a gaseous effluent release
- Steam leak action
- Routine radiological surveys
- Health Physics technician training
- Radiation work permits
- Startup test program
- Radioactive spill

3.0 Status of Previously Identified Items

3.1 (Open) Inspector Follow-up Item (352/84-66-06):

Collect and isotopically analyze PASS samples when sufficient activity levels are present. The licensee sampled the "A" RHR pathway from the PASS for a comparison with the Normal Sample Station. The inspector noted the comparison was conducted four times between 50 to 69 percent power level. The licensee stated that the PASS sample loops decreased the concentration differences from a factor of twenty-five to a factor of two. The licensee stated that continued sample comparisons to confirm that the PASS and normal sampling capabilities are within a factor of two will be performed. The inspector stated this action would be reviewed during a future inspection when radioactivity levels are sufficiently high to reduce analytical uncertainties.

4.0 Organization and Staffing

The organization and staffing of the health physics function was reviewed against criteria contained in:

- Technical Specification 6.2 - Organization
- Technical Specification 6.3 - Unit Staff Qualifications
- ANSI/ANS 3.1 - 1978, "Selection, Qualification, and Training of Personnel for Nuclear Power Plants"
- Reg Guide 1.8, "Personnel Selection and Training"
- ANSI/ANS 3.1 - 1978, "Administrative Control and Quality Assurance for the Operational Phase of Nuclear Power Plants."

The licensees performance relative to these criteria was determined from discussions with the Senior Health Physicist, and a review of position descriptions and personnel resumes.

The licensee has created two new superintendent positions. The Superintendent - Services is now responsible for three departments: Maintenance, Health Physics, and Outage Planning. Within the Health Physics Department, the ALARA Physicist has been moved under the Applied Health Physics section in order to balance the work load of the line supervisors. The licensee stated that these changes have been discussed with NRR and were made in an effort to enhance the various programs through increased management oversight.

Within the scope of this review, no violation was observed.

5.0 Gaseous Effluent Release

The licensees action with respect to a minor and unexpected release of gaseous effluent was reviewed against criteria contained in:

- Technical Specification 3/4.11.2 "Gaseous Effluents"
- Technical Specification 4.11.2.6.1 and 4.11.2.6.2 "Radioactive Effluents; Main Condenser"
- Station Procedure ST-6-104-880-0 "Gaseous Effluent Dose Rate Determination"
- Station Procedure EP-315 "Calculation of Offsite Doses During a (Potential) Radiological Emergency Using RMMS in the Manual Mode"

The action taken was determined by: interviews with the Support Health Physicist, Special Projects HP, Chemistry Supervisor, Count Room Chemist, and cognizant Test Engineer; a review of dose calculations; and a review of the operation of the RMMS monitoring system.

On January 2, 1986 during a controlled shutdown of the plant an in-rush of air into the turbine condenser occurred apparently as a result of cracked bellows in the cross-around piping. This air inleakage was pumped into the off gas system causing a pressure surge. The surge blew out a water seal in the radiation monitoring system and opened up a direct vent path through 1/2 inch sample piping to the north exhaust stack. The licensee's preliminary data indicates that the gaseous release rate in the stack peaked at 178.6 μCi per second and lasted less than 45 minutes. The technical specification limits for the release were not exceeded.

The licensee stated that the loop seal design will be evaluated to prevent a recurrence and that operations personnel were briefed regarding the problems that occurred. The chemistry technicians reported some delay in obtaining a grab sample for analysis due to locked security doors at the access to the north stack sample station. The licensee stated that the security controls to this area will be revised to allow expedited access for the technicians. This matter will be reviewed in a future inspection. (86-02-01)

6.0 Condenser Bay Steam Leak

After testing and instrumentation adjustments the plant was started and brought to full power on about December 28, 1985. The operators noted a steam flow versus power output mismatch indicating a loss of about 300,000 lbs/hr of the steam flow. On about January 2, 1986 it was determined that a steam relief valve on the cross-around piping was failed open and dumping low pressure steam into the condensers. At this time leaks developed from cracks in the expansion bellows used at the relief pipe ends, releasing steam into the condenser bay area. The steam condensed on the cold walls of the building with a small amount, estimated as less than 100 gallons permeating to the outside wall. The radioactivity in the condensed steam was measured by the licensee and found to be predominately N^{13} and F^{18}. The concentration of the F^{18} activity was 4×10^{-6} μCi/ml which is half of the limit for water provided in 10 CFR 20 Appendix B. The licensee concluded that there was no radiological hazards to personnel as a result of the steam leak. The low level radioactivity quickly dissipated due to the short half lives of the isotopes involved.

Within the scope of this review, no violations were observed. However, the inspector noted that the licensee does not have a procedure to capture the facts relative to potential radiological incidents and provide for a timely management review of these incidents. The licensee stated that there is ongoing management review of all significant events and that a procedure for documenting the events will be issued in February 1986. This matter will be reviewed in a future inspection . (86-02-02)

7.0 Routine Radiological Surveys

The licensees program for the conduct of routine radiological surveys was reviewed against criteria contained in:

- Technical Specifications 6.11, "Radiation Protection Program"
- 10 CFR 20.105, "Permissible Levels of Radiation in Unrestricted Area"
- 10 CFR 20.201, "Surveys"
- 10 CFR 20.203, "Caution signs, labels, signals and controls"
- 10 CFR 20.206, "Instruction of personnel"
- 10 CFR 20.401, "Records of surveys, radiation monitoring and disposal"
- Regulatory Guide 8.2, "Guide for Administrative Practices in Radiation Monitoring"
- IE Notice 84-82: Guidance for Posting Radiation Areas
- Station Procedures HP200, HP210, HP211, HP213, and HP215

The licensee's performance relative to these criteria was determined by:

- Discussion with the Health Physics Supervisor, Applied Health Physicist, and HP technicians,

- A review of completed radiation surveys and survey schedules,
- Observation of postings in selected plant areas,
- A review of the qualifications of technician performing the surveys.

Within the scope of this review, no violations were observed. In a few instances, the licensee has used only a three bladed magenta on yellow radiation symbol with no added wording posted on the door to a locked room. Technicians stated that this was done whenever the radiological conditions were expected to change. At the time of inspection these areas did not constitute Radiation or high Radiation Areas. The license was advised that IE Notice 84-82 states that postings should provide adequate information to workers to allow exposures to be minimized. The licensee stated that in the future all signs will follow generally accepted industry practice and regulatory requirements. In addition, permanent signs will be used whenever practicable. This matter will be reviewed in a future inspection. (86-02-03)

8.0 Health Physics Technician Training

The training and qualification program for Health Physics technician was reviewed against criteria contained in:

- Technical Specification 6.3, "Unit Staff Qualifications"
- Technical Specification 6.4, "Training"
- Technical Specification 6.10.3, "Record Retention"
- ANSI/ANS 3.1-1978, "Selection, Qualification, and Training of Personnel for Nuclear Power Plants"
- Station Procedure HP-100, "Health Physics Technician Selection, Training and Qualification"

The status of the licensees program was determined by:

- Interviews with the site and corporate Training Coordinators,
- Review of the "Nuclear Training Catalog", schedules, lesson plans and tests,
- Review of instructor training manuals and certifications,
- Review of selected qualification folders.

The licensee's training and qualification program for HP technicians is in various stages of development. The material for the Assistant Technician (AT), which is the first of six levels of progression, has been completed in draft form. The lesson plans and tests for the remaining levels in the stepwise qualification process will be developed as required. The licensee is coordinating this program with the Peach Bottom station.

Within the scope of this review, no violations were observed. The inspector noted examples of licensee strengths in this program. The lesson plans for AT level training were particularly thorough and technically sound. In addition, the licensee has tested the senior level qualified technicians hired at Peach Bottom, determined areas of weakness, and established a remedial training program for these technicians. The licensee indicted that INPO accreditation for the training courses is being sought.

9.0 Radiation Work Permits

The licensees implementation of procedure HP-310 "Radiation Work Permits" was reviewed by discussions with the Health Physics Supervisor, accompanying technicians during a pre-job survey, and an inspection of records. There was no work in progress that required an RWP. The licensee stated that all work, including work by any contractor, is assigned a Maintenance Request (MRF) number. A computer is then used to record all important information regarding any work, including the requirement for an RWP. This system was adopted, with some modification, from the Peach Bottom station. The inspector concluded that the MRF system can provide adequate control of work. However, the low levels of plant contamination at the present time do not require frequent use of RWP's. This area will be reviewed again in a future inspection.

10.0 Start-up Tests: Chemical and Radiochemical

The inspector reviewed licensee Start-up Test results for chemical and radiochemical tests and gaseous radioactive waste system tests. The following specific Start-up Tests were reviewed: STP 1.2, Power Ascension Chemistry/Radiochemistry; STP 1.3, Gaseous Effluents; and STP 34.1, Offgas Performance. The Start-up Test results were reviewed against the acceptance criteria contained in the Start-up Test procedures.

The Start-up Tests results review indicated that the licensee established reactor water quality parameters that met the Technical Specification requirements, and demonstrated the ability to maintain the specifications during operation up to 80% reactor power. Also, the Start-up Test results indicated that the offgas system, which had been tested through the 100% reactor power level met the performance specifications stated in the FSAR, and gaseous radioactive effluent releases were within Technical Specification limits.

The inspector noted that Start-up Test 34.1, Offgas Performance, performed at both the 65-80 percent power levels, and the 90-100 percent power levels contained both arithmetical and transcription errors. These results had not been reviewed and approved. These errors were discussed with license and licensee contractor personnel. In addition, Start-up Test 1.3, Gaseous Effluents, performed at the 45-55% power level contained an error, in that the improper plant vent monitor reading was recorded in Appendix A of the test. This test, however, was reviewed by PORC and approved. The licensee stated that all three tests would be corrected. The inspectors noted that with the necessary corrections the tests still met all acceptance criteria. The inspector stated that the corrections would be reviewed during a subsequent inspection. (352/86-02-04)

The inspector also witnessed a demonstration of the licensee's computer system for maintaining and trending chemistry data. This system was examined during a previous inspection (50-352/85-23 conducted April 23-26, 1985) of this area but at that time the system was in the development stages. During this demonstration, graphs of various chemical parameter versus time were shown to the inspector as well as the actual data files. Although the system is not completely implemented, it appears that the licensee has developed a chemistry data base system which will contribute to the licensee's ability to meet plant system chemistry parameters.

The inspector had no further questions in this area. No violations were identified.

Start-up Testing: Radiation Surveys

Documents Reviewed

- Final Safety Analysis Report (FSAR), Chapter 14, "Initial Test Program"
- Start-up Test Procedure STP 2.0, Revision 1, "Radiation Measurements -Main Body", dated September 13, 1984
- Start-up Test Procedure STP 2.1-6, Revision 1, "Start-up Radiation Surveys-Prior to Fuel Load", dated December 27, 1985
- ANSI/ANS-6.3.1, 1980, "Program for Testing Radiation Shields in Light Water Reactors (LWR)"

Review of the test procedures and test data indicated that the licensee was conducting start-up radiation surveys in accordance with FSAR commitments and procedural requirements. There were no unexpected levels of radiation except for one location. This reading was 34 mr/hr (Zone III). The licensee plans on resolving this test result by redesignating the area as a Zone II. The licensee PORC review of the test results was not completed.

11.0 Radioactive Spill

The inspector discussed a spill which occurred at the condensate sampling station on January 8, 1986. The drain lines from the sample sink at this sampling station became inoperable when the plant was shut down and drain line vacuum was lost. The sample sink overflowed into a floor drain which was pumped to the onsite holding pond. The holding pond is discharged to the Schuylkill River. Analysis of the liquid in the sample sink indicated only Co-58 at a concentration of 4.03 E-6 µCi/ml. This concentration was less than the unrestricted area MPC of 9 E-5 µCi/ml for Co-58 prior to dilution in the holding pond. A sample of the holding pond indicated less than detectable levels of Co-58. The licensee stated that evaluations were being performed in order to ensure operation of the sample station drain lines when vacuum was lost. The inspector stated that this area would be reviewed during a subsequent inspection. (352/86-02-05)

The inspector had no further questions in this area. No violations were identified.

12.0 Exit Interview

The inspector met with licensee representatives at the conclusion of the inspection on January 10, 1986. The scope and findings of the inspection were discussed at that time. At no time was written material provided to the licensee by the NRC inspector.

QUESTION 421.10 (Section 7.7)

The analyses reported in Chapter 15 of the FSAR are intended to demonstrate the adequacy of safety systems in mitigating anticipated operational occurrences and accidents.

Based on the conservative assumptions made in defining these "design bases" events and the detailed review of the analyses by the staff, it is likely that they adequately bound the consequences of single control failures. To provide assurance that the design basis event analysis for Limerick adequately bounds other more fundamental credible failures, provide the following:

(1) Identify those control systems whose failure or malfunction could seriously impact plant safety.

(2) Indicate which, if any, of the control systems identified in (1) receive power from common power sources. The power sources considered should include all power sources whose failure or malfunction could lead to failure or malfunction of more than one control system and should extend to the effects of cascading power losses due to the failure of higher level distribution panels and load centers.

(3) Indicate which, if any, of the control systems identified in (1) receive input signals from common sensors. The sensors considered should include common Taps, hydraulic headers and impulse lines feeding pressure, temperature, level or other signals to two or more control systems.

(4) Provide justification that any malfunctions of the control systems identified in (2) and (3) resulting from failures or malfunctions of the applicable common power source or sensor including hydraulic components are bounded by the analyses in Chapter 15 and would not require action or response beyond the capability of operators or safety systems.

RESPONSE

The requested information is provided in the Control Systems Failures Evaluation Report and the Common Sensor Failure Evaluation Report which were transmitted by letter from J. S. Kemper to A. Schwencer dated December 14, 1983.

QUESTION 421.11 (Section 7.7)

Section 7.7.1.1.3.1.5 of the FSAR indicates that the RPV pressure and water level instruments use the same instrument lines. Identify all other cases where instrument sensors or transmitters supplying information to more than one protection channel are located in a common instrument line or connected to a common instrument tap. Verify that a single failure in a common instrument line or tap (such as break or blockage) cannot defeat required protection system redundancy. Identify where instrument sensors or transmitters supplying information to both a protection channel and one or more control channels are located in a common instrument line or connected to a common instrument tap. Verify that a single failure in a common instrument line or tap cannot defeat required separation between control and protection.

RESPONSE

The requested information is provided in the Common Sensor Failure Evaluation Report which was transmitted by letter to J. S. Kemper to A. Schwencer dated December 14, 1983.

PHILADELPHIA ELECTRIC COMPANY

2301 MARKET STREET

P.O. BOX 8699

PHILADELPHIA, PA. 19101

(215) 841-4000

January 21, 1986

Docket No. 50-352

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: Licensee Event Report
Limerick Generating Station - Unit 1

This LER deals with excessive leakage identified during local leak rate testing of the drywell spray outboard isolation valve, HV-51-1F016A.

Reference:	Docket No. 50-352
Report Number:	85-102
Revision Number:	00
Event Date:	December 18, 1985
Report Date:	January 21, 1986
Facility:	Limerick Generating Station P.O. Box A, Sanatoga, PA 19464

This LER is being voluntarily submitted based on the possible significance of the event.

Very truly yours,

W. T. Ullrich
Superintendent
Nuclear Generation Division

cc: Dr. Thomas E. Murley, Administrator, Region I, USNRC
E. M. Kelly, Senior Resident Site Inspector
See Service List

cc: Troy B. Conner, Jr., Esq.
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Atomic Safety & Licensing Appeal Board
Atomic Safety & Licensing Board Panel
Docket & Service Section (3 Copies)
E. M. Kelly
Timothy R. S. Campbell

September, 1985

NRC Form 366 (9-83)

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/85

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Limerick Generating Station - Unit 1	0 5 0 0 0 3 5 2	1 OF 0 3

TITLE (4): Excessive Leakage of Containment Spray Valve, HV-51-1F016A

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAMES	DOCKET NUMBER(S)
1 2	1 8	8 5	8 5	1 0 2	0 0	0 1	2 1	8 6		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 0 7 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)

[X] OTHER (Specify in Abstract below and in Text, NRC Form 366A)

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER
John C. Nagle, Senior Engineer, Licensing Section	2 1 5 8 4 1 - 5 1 8 4

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS		CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	
X	B O	I S V	L 2 0 0	Y							

SUPPLEMENTAL REPORT EXPECTED (14)

YES (If yes, complete EXPECTED SUBMISSION DATE) | NO

EXPECTED SUBMISSION DATE (15): MONTH 0 7, DAY 1 5, YEAR 8 6

ABSTRACT (Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines) (16)

Abstract: 85-102

On December 18, 1985, with Unit No. 1 at power operation and at 69.9 percent power, while performing local leak rate testing, containment spray header outboard isolation valve HV-51-1F016A, was identified as not being fully closed as evidenced by having excessive leakage. While pressurizing between the inboard and the outboard motor operated isolation valves, the test pressure could not be achieved until the HV-51-1F016A valve was manually closed using its handwheel. The valve was declared inoperable and administratively secured in the fully closed position. The cause of this event was believed to be either improper operation of the valve operator or excessive internal valve friction. Further investigation will be performed.

NRC Form 366 (9-83)

Unit Condition Prior to the Event:

Mode 1 (Power Operation)
Reactor Power 69.9%
Power Ascension Testing

Description of the Event:

On December 18, 1985, while performing local leak rate testing, drywell spray header outboard isolation valve, HV-51-1F016A, was identified as having excessive leakage. While pressurizing between the inboard motor operated isolation valve, HV-51-1F021A, and the outboard motor operated isolation valve, HV-51-1F016A, the test pressure of 44 psig (peak accident pressure) could not be achieved until the HV-51-1F016A valve was manually closed uing its handwheel. The HV-51-1F016A valve was declared i perable and the local leak rate test was completed successfully with the valve in its manually closed position. Plant operation is continuing with the HV-51-1F016A valve secured in a closed position per Technical Specification requirements.

The EIIS code for the affected system is BO; for the affected component is ISV.

Consequences of the Event:

Since the local leak rate test results were acceptable with the 16A valve in the manually closed position, primary containment integrity was maintained with the HV-51-1F021A. The HV-51-1F016A and HV-51-1F021A valves are normally closed and the previous local leak rate test results of the containment penetration were acceptable. In the case of high drywell pressure as would be experienced in a Loss of Coolant Accident, the HV-51-1F016A and HV-51-1F021A valves could be opened to provide containment spray along with the redundant "B" loop of containment spray through the HV-51-1F016B and HV-51-1F021B valves.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Limerick Generating Station Unit 1	0 5 0 0 0 3 5 2	8 5	— 1 0 2	— 0 0	0 3 OF 0 3

TEXT (If more space is required, use additional NRC Form 366A's) (17)

Cause of the Event:

The cause of this event is either improper operation of the Limitorque operator causing the motor to stop due to torque prior to full closure of the valve or high internal valve friction. Troubleshooting has revealed that excessive grease in the torque sensing mechanism of the Limitorque operator is possibly causing improper functioning of the valve. Further investigation will be performed to resolve this problem.

Corrective Actions:

A supervisory block was applied to the hand switch in the control room, the motor control center and the manual handwheel for the HV-51-1F016A valve to secure it in the full closed position and de-energized during plant operation. The valve in this state is considered manual and will not be unblocked or considered an operable motor operated valve until successful completion of the local leak rate test. During the next plant shutdown of sufficient duration, maintenance will be performed on the valve operator which will include cutting a slot in the torque limit sleeve of the spring pack used to sense motor torque. This will allow the release of any excess grease and prevent possible improper valve operation. At that time, a local leak rate test will be performed to verify no further problems.

Previous Similar Occurrences

None.

PHILADELPHIA ELECTRIC COMPANY

2301 MARKET STREET

P.O. BOX 8699

PHILADELPHIA, PA. 19101

(215) 841-5020

M. J. COONEY
MANAGER
NUCLEAR PRODUCTION
ELECTRIC PRODUCTION DEPARTMENT

January 29, 1986

Docket No. 50-352

Mr. Robert Bernero, Director
Division of Boiling Water Reactor Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Bernero:

The following information is provided in response to a Request for Additional Information (RAI) from Mr. R. E. Martin, Limerick Project Manager, of your staff, in support of a December 18, 1985 Request for Amendment to the Limerick Operating License and Temporary Exemption to the requirements of Appendix J to 10 CFR 50.

This request was directed towards supplying information relative to industry experience with valves similar to those which were the subject of the Amendment Request. In order to support this RAI, a multi-faceted program has been undertaken which includes the following: 1) Review of Nuclear Plant Reliability Data systems; 2) Contact with other utilities identified as having similar valves; 3) Contact with valve manufacturers of the specific valves; and 4) Review of general experience with testing of similar valves at Peach Bottom Atomic Power Station (PBAPS).

Table 1 (attached) addresses item 1 above. Items 2 and 3 remain under investigation. In response to item 4, general experience with similar valves at PBAPS indicates that valve leakage appears to be related to type of service and time in service. Valves which are used in non-modulating applications, such as those which are the subject of this amendment request, tend not to have problems meeting leakage criteria.

Table 2 (attached) is a compilation of information regarding the valves for which temporary relief of the testing requirement is sought.

The information gathered thus far has revealed nothing which would alter or affect the conclusions contained within our application.

Should you have any questions or require additional information, please do not hesitate to contact us.

Very truly yours,

Attachments

cc: Dr. Thomas E. Murley, Administrator, Region I, USNRC
E. M. Kelly, Senior Resident Site Inspector
See Attached Service List

cc: Troy B. Conner, Jr., Esq.
Ann P. Hodgdon, Esq.
Mr. Frank R. Romano
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Jay M. Gutierrez, Esq.
Atomic Safety & Licensing Appeal Board
Atomic Safety & Licensing Board Panel
Docket & Service Section
E. M. Kelly
Timothy R. S. Campbell

Table 1 - Explanation

Results of review of industry experience conducted through Nuclear Plant Reliability Data System

Columns 1, 2, 3, 4

Manufacturer, Model Number, Size and Type of valves which are the subject of the amendment request.

Column 5 - Similar Number Identified

Number of valves identified upon interrogation of NPRD data base. Search was based upon manufacturer, type and size range (e.g. Velan, gate, 2"-6"). Computer generated listing was manually sorted to identify similar model numbers. If no similar model numbers were identified, the number reported represents the numbers of all valves of same manufacturer, type and specific size.

Column 6

NPRD failure reports were obtained for all valves identified and reported in Column 5. These reports were reviewed and those failures which were relevant to leakage rates and isolation function are tabulated. It is notable that the reported failures concern valves which have been in service for significant periods and were reported to NPRD as "wear-out".

Column 7

Identifies those totals which are reported for all valves of same size when similar model numbers could not be identified.

TABLE 1

NPRDS REVIEW
OF
SIMILAR VALVES

MANUFACTURER	MODEL	SIZE	TYPE	SIMILAR NUMBER IDENTIFIED	MEANINGFUL FAILURES REPORTED	COMMENTS
Velan	B12-00054B-02WN B10-00054B-02WN	3"	Gate	30	0	
Crane	Cat. #47XUF	8"	Gate	198	0	
Rockwell	1-3624F316LMT 1 1/2-3624F316LMMT	1" 1 1/2"	Globe Globe	492	28	
Atwood & Morrill	13673-02-03-05-06 13673-01-04	12"	CK	0(61*)	6*	*All Models.
Anchor/Darling	SJO-2171-52	8"	Gate	0(337*)	4*	*All Models.
	SJO-2159-15 SJO-2159-16 SJO-5348-06	6"	Globe	0(46*)	1*	*All Models.
	SJO-5348-07	12"	Gate	0(226*)	1*	*All Models.
Borg Warner	Part No. 77940	1"	CK	0(21*)	0	
Circle Seal	NV02-14	1"	CK	0(14*)	2*	*All Models.

TABLE 2

Penetration Number and Description	Valve Numbers (Size - Inches)	Test Medium	Frequency of Valve Operation	Previous Test Results	Process Fluid	Manufacturer	Valve Type	Model Number
X-3D Instrument Gas Supply	HV-59-151B (1) 59-1112 (1)	Air	N.O. Stroked Quart Check Stroked Refuel	7.54 SCCM 1.45 SCCM	PCIG or Instrument Air	Rockwell Circle Seal	Globe Check	1-3624 F316LM WV02-14
X-13A 'A' RHR Shutdown Cooling Return	HV-51-1F050A (12) HV-51-151A (1.5)	Air/ Water	Check Stroked Refuel N.C. Not Stroked	0.1 GPM* 0.1 GPM*	Reactor Demineralized Water	Atwood & Morrill Rockwell	Check Globe	13673-01-04 1 1/2-3624F316
X-13B 'B' RHR Shutdown Cooling Return	HV-51-1F050B (12) HV-51-151B (15)	Air/ Water	Check Stroked Refuel N.C. Not Stroked	0.0 GPM* 0.0 GPM*	Reactor Demineralized Water	Atwood & Morrill Rockwell	Check Globe	13673-01-04 1 1/2-3624F316 LM
X-14 Reactor Water Cleanup Supply	HV-44-1F001 (6) HV-44-1F004 (6)	Air	N.O. Stroked Quart N.O. Stroked Quart	24.0 SCCM 485.75 SCCM	Reactor Demineralized Water	Anchor/Darling Anchor/Darling	Globe Globe	SJO-2159-15 SJO-2159-16
X-23 Reactor Enclosure Cooling Supply	HV-13-106 (4) HV-13-108 (3) HV-13-109 (3)	Air	N.O. Stroked Refuel N.O. Stroked Refuel N.C. Not Stroked	121.8 SCCM 23.25 SCCM* 23.25 SCCM*	React. Encl. Cooling Water Demin. Water	Velan Velan Velan	Gate Gate Gate	B12-00054B-02WN B10-00054B-02WN B10-00054B-02WN
X-24 Reactor Enclosure Cooling Return	HV-13-107 (4) HV-13-110 (3) HV-13-111 (3)	Air	N.O. Stroked Refuel N.C. Not Stroked N.O. Stroked Refuel	5.2 SCCM 3.6 SCCM* 3.6 SCCM*	React. Encl. Cooling Water Demin. Water	Velan Velan Velan	Gate Gate Gate	B12-00054B-02WN B10-00054B-02WN B10-00054B-02WN
X-45A 'A' RHR LPCI	HV-51-1F041A (12) HV-51-1F017A (12) HV-51-142A (1.5)	Air/ Water	Check Stroked Refuel N.C. Stroked Refuel N.C. Not Stroked	0.0198 GPM* 84.75 SCCM 0.0198 GPM*	Suppression Pool Water	Atwood & Morrill Anchor/Darling Rockwell	Check Gate Globe	13673-02-03-05-06 SJO-5348-07 1-3624F316LMMT
X-45C 'C' RHR LPCI	HV-51-1F041C (12) HV-51-1F017C (12) HV-51-142C (1.5)	Air/ Water	Check Stroked Refuel N.C. Stroked Refuel N.C. Not Stroked	0.002 GPM * 148. SCCM 0.002 GPM *	Suppression Pool Water	Atwood & Morrill Anchor/Darling Rockwell	Check Gate Globe	13673-02-03-05-06 SJO-5348-07 1-3624F316LMMT
X-45D 'D' RHR LPCI	HV-51-1F041D (12) HV-51-1F017D (12) HV-51-142D (1.5)	Air/ Water	Check Stroked Refuel N.C. Stroked Refuel N.C. Not Stroked	0.0828 GPM* 976.8 SCCM 0.0828 GPM*	Suppression Pool Water	Atwood & Morrill Anchor/Darling Rockwell	Check Gate Globe	13673-02-03-05-06 SJO-5348-07 1-3624F316LMMT
X-53 Drywell Chilled Water Supply	HV-87-120A (8) HV-87-125A (8) HV-87-128 (8)	Air	N.O. Stroked Quart N.C. Stroked Quart N.O. Stroked Quart	170.5 SCCM* 170.5 SCCM* 32.25 SCCM	Demineralized Water	Crane Crane Anchor/Darling	Gate Gate Gate	Cat. No. 47XUF Cat. No. 47XUF SJO-2171-52

TABLE 2

Penetration Number and Description	Valve Numbers (Size - Inches)	Test Medium	Frequency of Valve Operation	Previous Test Results	Process Fluid	Manufacturer	Valve Type	Model Numbe
X-54 Drywell Chilled Water Return	HV-87-121A (8) HV-87-124A (8) HV-87-129 (8)	Air	N.O. Stroked Quart N.C. Stroked Quart N.O. Stroked Quart	556.2 SCCM* 556.2 SCCM* 97.2 SCCM	Demineralized Water	Crane Crane Anchor/Darling	Gate Gate Gate	Cat. No. 47XUF Cat. No. 47XUF SJO-2171-52
X-55 Drywell Chilled Water Supply	HV-87-120B (8) HV-87-122 (8) HV-87-125B (8)	Air	N.O. Stroked Quart N.C. Stroked Quart N.C. Stroked Quart	656.5 SCCM* 11.4 SCCM 656.5 SCCM*	Demineralized Water	Crane Anchor/Darling Crane	Gate Gate Gate	Cat. No. 47XUF SJO-2171-52 Cat. No. 47XUF
X-56 Drywell Chilled Water Return	HV-87-121B (8) HV-87-123 (8) HV-87-124B (8)	Air	N.O. Stroked Quart N.C. Stroked Quart N.C. Stroked Quart	302.6 SCCM* 35.5 SCCM 302.6 SCCM*	Demineralized Water	Crane Anchor/Darling Crane	Gate Gate Gate	Cat. No. 47XUF SJO-2171-52 Cat. No. 47XUF
X-61B 'B' Recirc. Pump Seal Purge	43-1004B (1)	Air	Check Stroked Refuel	75.9 SCCM	CRD or Reactor Demin. Water	BORG Warner	Check	Part No. 77940
X-205A Suppression Pool Spray	HV-51-1F027A (6)	Air	N.C. Stroked Quart	2.25 SCCM	Suppression Pool Water or Air	Anchor/Darling	Globe	SJO-5348-06
			N.O. Normally Open N.C. Normally Closed	*Valves tested together. Leakage assigned to both.				
				Current total of type C test 22,000 SCCM				

NUREG-1149

Technical Specifications

Limerick Generating Station, Unit No. 1

Docket No. 50-352

Appendix "A" to
License No. NPF-39

Issued by the
U.S. Nuclear Regulatory
Commission

Office of Nuclear Reactor Regulation

June 1985

REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.3.2 Reactor coolant system leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE.

b. 5 gpm UNIDENTIFIED LEAKAGE.

c. 30 gpm total leakage.

d. 25 gpm total leakage averaged over any 24-hour period.

e. 1 gpm leakage at a reactor coolant system pressure of 950 ±10 psig from any reactor coolant system pressure isolation valve specified in Table 3.4.3.2-1.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. With any reactor coolant system leakage greater than the limits in b. and/or c., above, reduce the leakage rate to within the limits within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With any reactor coolant system pressure isolation valve leakage greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least one other closed manual, deactivated automatic, or check* valves, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

d. With one or more of the high/low pressure interface valve leakage pressure monitors shown in Table 3.4.3.2-1 inoperable, restore the inoperable monitor(s) to OPERABLE status within 7 days or verify the pressure to be less than the alarm setpoint at least once per 12 hours; restore the inoperable monitor(s) to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

*Which have been verified not to exceed the allowable leakage limit at the last refueling outage or after the last time the valve was disturbed, whichever is more recent.

4.4.3.2.1 The reactor coolant system leakage shall be demonstrated to be within each of the above limits by:

a. Monitoring the primary containment atmospheric gaseous radioactivity at least once per 12 hours (not a means of quantifying leakage),

b. Monitoring the drywell floor drain sump and drywell equipment drain tank flow rate at least once per 12 hours,

c. Monitoring the drywell unit coolers condensate flow rate at least once per 12 hours,

d. Monitoring the primary containment pressure at least once per 12 hours (not a means of quantifying leakage),

e. Monitoring the reactor vessel head flange leak detection system at least once per 24 hours, and

f. Monitoring the primary containment temperature at least once per 24 hours (not a means of quantifying leakage).

4.4.3.2.2 Each reactor coolant system pressure isolation valve specified in Table 3.4.3.2-1 shall be demonstrated OPERABLE by leak testing pursuant to Specification 4.0.5 and verifying the leakage of each valve to be within the specified limit:

a. At least once per 18 months, and

b. Prior to returning the valve to service following maintenance, repair or replacement work on the valve which could affect its leakage rate.

The provisions of Specification 4.0.4 are not applicable for entry into OPERATIONAL CONDITION 3.

4.4.3.2.3 The high/low pressure interface valve leakage pressure monitors shall be demonstrated OPERABLE with alarm setpoints set less than the allowable values in Table 3.4.3.2-1 by performance of a:

a. CHANNEL FUNCTIONAL TEST at least once per 31 days, and

b. CHANNEL CALIBRATION at least once per 18 months.

7.6.1.1.9.4 NSE-RMS Testability

Built-in radioactive check sources for simulating mid-range radiation levels are provided for each channel for test purposes. These tests are conducted by an operator stationed in the auxiliary equipment room. Remote-controlled purge capability is provided. The operability of each monitoring channel can be routinely verified by comparing the outputs of the two monitoring systems during power operation.

7.6.1.1.9.5 NSE-RMS Environmental Considerations

The wide-range accident monitor has been designed and qualified to meet environmental conditions under all modes of plant operation, including accidents. The normal-range monitor is designed to withstand the normal service environment.

7.6.1.1.9.6 NSE-RMS Operational Considerations

Annunciation, computation, and recording capabilities are provided for this system. The equipment is located in an area where the radiation environment is sufficiently low to afford personnel access over the range of plant operating conditions. However, the instrumentation is designed for remote operation and control as well as data retrieval.

7.6.1.2 High-Pressure/Low-Pressure Systems Interlocks (HPLPSI) - Instrumentation and Controls

7.6.1.2.1 HPLPSI Function Identification

The low-pressure systems that interface with the reactor coolant pressure boundary (RCPB), and the instrumentation that protects them from overpressurization, are discussed in this section.

7.6.1.2.2 HPLPSI Power Sources

The power for the interlocks is provided from the essential power supplies for the associated systems except for the RHR steam condensing mode steam supply line valves.

7.6.1.2.3 HPLPSI Equipment Design

At least two isolation valves are provided in series in each line, except for the RHR high-pressure/low-pressure interface on the steam condensing mode steam supply line which has a motor-operated block valve in series with a pressure reducing valve and a relief valve on the low pressure side.

The following high-pressure/low-pressure interlock equipment is provided:

Interlocked Process Line	Type	Valve	Parameter Sensed	Purpose
RHR shutdown cooling supply	MO MO	HV51-F009 HV51-F008	Reactor pressure	Prevents valve opening until reactor pressure is low
RHR shutdown Cooling return	Check	51-F050A,B	N/A	N/A
	MO	HV51-F015A,B	Reactor pressure	Prevents valve opening until reactor pressure is low
	AO	HV51-151A,B	Note 1	Note 1
RHR head spray	MO MO	HV51-F022 HV51-F023	Reactor pressure	Prevents valve opening until reactor pressure is low
RHR LPCI line	AO	HV51-142A,B, C,D	Note 1	Note 1
	Check	51-F041A,B, C, D	N/A	N/A
	MO	HV51-F017A,B, C, D	Differential pressure across valve	Prevents valve opening until differential pressure is low
RHR steam condensing mode steam supply line	MO MO AO	HV51-153A,B HV51-F052A,B HV51-F051A,B	Note 2	Steam supply line, Block valves, Pressure reducing valve
	MO	HVC51-154A,B	Note 3	
CS system	Check	HV52-F006A,B	N/A	N/A
	Check	HV52-108	N/A	N/A
	MO MO MO	HV52-F005 HV52-F004 HV52-F037	Reactor pressure	Prevents valve opening until reactor pressure is low
	AO	HV52 F039A,B	Note 1	Note 1

MSIV-LCS	MO	HV40-F001B,F,P,K	Prevents system initiation until reactor pressure is low
	MO	HV40-F002B,F,K,P	
	MO	HV40-F003B,F,K,P	
	MO	HV40-F006	
	MO	HV40-F007	
	MO	HV40-F008	Note 4
	MO	HV40-F009	

Note 1: No parameter sensed because the valves are opened solely by remote momentary pushbuttons, to equalize pressure across the check valve discs to permit testing of the opening of the check valves.

Note 2: No parameter sensed for HPLPSI. Line pressure is sensed to position HV51-1F051A,B to maintain RHR heat exchanger shell pressure. Low pressure piping overpressure protection is provided by a relief valve on the low pressure piping.

Note 3: HV-C-51-154A,B is locked in the closed position and electrical power to the valve has been disconnected.

Note 4: Electrical power for valves HV40-F001, F002, F003B, F, K, and P and HV40-F006, F007, F008, and F009 is removed during plant operation by locking the MCC breakers in the open position.

7.6.1.2.3.1 HPLPSI Circuit Description

The RHR shutdown cooling suction valves from the recirculation line have independent interlocks to prevent the valves from opening when the reactor pressure is above the RHR system design pressure. These valves also receive a signal to close when reactor pressure is above the RHR system pressure.

The RHR system head spray motor-operated valves have independent interlocks to prevent opening whenever the reactor pressure is above the system design pressure, and to automatically close whenever the reactor pressure exceeds the system design pressure.

The RHR system shutdown cooling discharge valves have two reactor pressure interlocks, both of which must be permissive to allow opening the valves. Each line has a remote testable check valve downstream of the discharge valve. The check valve position can be confirmed at any time.

The RHR system low pressure coolant injection subsystem (LPCI) injection valves open when differential pressure across the valves is low. There is a remote testable check valve downstream of the injection valve in each loop.

The CS system injection valves open when reactor pressure decreases below the system design pressure. There is a remote testable check valve downstream of each injection valve. There is an additional check valve downstream of the injection valve on loop B.

7.6.1.2.3.2 HPLPSI Logic and Sequencing

The RHR shutdown cooling valves and the RHR head spray valves are interlocked by reactor pressure in a two-out-of-two logic. In all other cases, the sensor inputs operate the interlocks without logic combination.

7.6.1.2.3.3 HPLPSI Bypasses and Interlocks

There are no bypasses or interlocks in the high-pressure/low-pressure interlocks.

7.6.1.2.3.4 HPLPSI Redundancy

Each process line has two valves in series that are redundant in ensuring the interlock except for the RHR steam condensing mode steam supply line as defined in Section 7.6.1.2.3. The RHR shutdown cooling suction valves and the RHR head spray valves have independent interlocks to prevent the valves from opening when the reactor pressure is above the system design pressure.

7.6.1.2.3.5 HPLPSI Actuated Devices

The motor-operated valves listed in Section 7.6.1.2.3 are the actuated devices.

7.6.1.2.3.6 HPLPSI Separation

Separation is maintained between redundant portions of the high-pressure/low-pressure interlocks by assigning the signals for the redundant electrically-controlled valves to separate electrical divisions. (Refer to Section 7.1.2.2.)

7.6.1.2.3.7 HPLPSI Testability

The actuated devices (except those valves kept closed by reactor pressure interlocks) can be tested during reactor operation. The sensors are tested during reactor operation in the same manner that engineered safety feature (ESF) sensors are tested. Refer to Section 7.3.1 for a discussion of testing ESF sensors.

7.6.1.2.4 HPLPSI Environmental Considerations

The instrumentation and controls for the high-pressure/low-pressure interlocks are qualified as Class IE equipment. The sensors are mounted on local instrument panels and the control circuitry is housed in panels in the auxiliary equipment room and the control room. Refer to Sections 3.10 and 3.11 for details of the qualification testing.

7.6.1.2.5 HPLPSI Operational Considerations

7.6.1.2.5.1 HPLPSI General Information

The high-pressure/low-pressure interlocks are strictly automatic. There is no manual actuation capability. If the operator initiates a low-pressure system, the interlocks prevent exposure of the low-pressure piping to high pressure.

7.6.1.2.5.2 HPLPSI Reactor Operator Information

The status of each valve providing the high-pressure/low-pressure boundary is indicated in the control room. The state of the reactor pressure and RHR injection valve differential pressure sensors is indicated in the control room.

7.6.1.2.5.3 HPLPSI Setpoints

The setpoints for HPLPSI are contained in Chapter 16, Technical Specifications.

7.6.1.3 <u>Leak Detection System (LDS) - Instrumentation and Controls</u>

The LDS consists of the following safety-related subsystems:

a. Main steam line leak detection subsystem

b. RCIC system leak detection subsystem

c. RWCU system leak detection subsystem

d. HPCI system leak detection subsystem

7.6.1.3.1 LDS Identification

This section discusses the instrumentation and controls associated with the safety-related portion of the leak detection system. The non-safety-related portion is described in Section 7.7.1.16. The LDS itself is discussed in Section 5.2.5.

The purpose of the leak detection system instrumentation and controls is to detect and provide the signals necessary to isolate leakage from the RCPB before predetermined limits are exceeded. Environmental conditions and qualification for the leak detection system are discussed in Sections 3.10 and 3.11. Seismic qualification of the main steam line break detection subsystem is discussed in Section 7.3.2.2.2.3.1.5.

QUESTION 421.50 (Section 7.6)

Section 7.6.1.2.3.6 of the FSAR indicates that for the high pressure low pressure system interlocks (HPLPSI), separation is maintained by assigning signals for electrically-controlled valves to separate electrical divisions. Discuss how the overall separation of the HPLPSI complies with the guidance provided in R.G. 1.75 without compromising systems in different divisions. This can be discussed in conjunction with item 421.21. Discuss the degree of conformance to the guidelines provided in ICSB, BTP-3, for the HPLPSI as implemented in your design.

RESPONSE

The redundant high pressure interlocked valves are powered by separate essential electrical power. The relay logic, which performs the function of preventing valve opening unless pressure is below setpoint, is the same essential power as that of the associated interlocked valve. The interlocked valves of the high pressure/low pressure system interlocks are as follows:

	VALVE	POWER
(RHR)	E11-F008	ESS 2
	E11-F009	ESS 1
	E11-F050A, B	ESS 1
	E11-F015A, B	ESS 2
	E11-F022	ESS 1
	E11-F023	ESS 2
	E11-F017A	ESS 1
	E11-F017B	ESS 2
	E11-F017C	ESS 3
	E11-F017D	ESS 4
(CS)	E21-F004A	ESS 1
	E21-F004B	ESS 2
	E21-F005	ESS 1
	E21-F037	ESS 2
(MSIV-LCS)	E32-F001B, F, P, K	ESS 2
	E32-F002B, F, P, K	ESS 2
	E32-F003B, F, P, K	ESS 2
	E32-F006	ESS 1
	E32-F007	ESS 1
	E32-F008	ESS 1
	E32-F009	ESS 1

The sensors that actuate the interlock logic (on pressure below setpoint) are on separate instrument lines and power such that no single failure can prevent core cooling. The electrical separation of the HPLPSI is consistent with the systems of which they comprise a part and represent no deviation from the intent of Regulatory Guide 1.75 as discussed in Section 8.1.6.1.14.

The interlocked valves of the high pressure/low pressure system interlocks (HPLPSI) meet Branch Technical Position 3 (BTP-3) in accordance with the following:

Two Motor-Operated Valves in Series (BTP-3), paragraph 3)

E11-F008 and E11-F009 (RHR shutdown cooling suction outboard and inboard valves, respectively) are two manually activated motor operated valves in series. Both valves are inhibited from opening and close automatically if primary system pressure is above setpoint. Reactor pressure is also indicated in the control room. The logic components for both valves are independent. Each valve control circuit requires two reactor low pressure permissives before valves can open; this results in a four-out-of-four logic to open the suction line. Removal of one signal (one-out-of-four logic) isolates the line. The pressure permissive components rely on the transmitter trip-unit combination which is testable from the control room.

Reactor pressure instrumentation used by the operator (via plant procedures) to initiate shutdown cooling is independent of the interlocks. Procedural controls ensure that the manual initiated shutdown cooling mode is not begun until the reactor pressure is below approximately 135 psig; this constitutes a safety factor of more than 3 times. (Low pressure systems are rated at approximately 475 psig).

Because of the foregoing additional safety design features, diversity of interlocks as suggested by BTP.3 paragraph 2 has not been implemented for Limerick. This is consistent with all other BWR testability (transmitter-trip unit) enhanced plants such as Grand Gulf.

E11-F023 and E11-F022 (RHR reactor head spray outboard and inboard valves, respectively) have an interlock description identical to the previous description for valves E11-F008 and E11-F009.

The valves of the MSIV leakage control system listed below have the following system of high/low pressure interlocks:

E32-F001B, F, P, K	MSIV inboard MOVs
E32-F002B, F, P, K	MSIV inboard MOVs
E32-F003B, F, P, K	MSIV Flow XMTR Bypass MOVs
E32-F006 and E32-F007	Outboard Bleed MOVs
E32-F008 and E32-F009	Outboard Depress MOVs

The inboard MSIV-LCS valves are interlocked by pressure permissives from the reactor pressure vessel and volume between the inboard and outboard MSIVs. The outboard set of MSIV-LCS valves are interlocked by the reactor pressure sensor and a pressure sensor in the volume outboard of the outboard MSIV. Motive power to these MSIV-LCS valves is removed during plant operation.

<u>Motor-Operated Valve in Series with an Air Operated Valve (Not Addressed in BTP-3)</u>

E11-F052A and E11-F052B (steam line MOVs) are motor operated valves in series with E11-F051A and E11-F051B (steam pressure reducing AOVs), respectively. These motor operated valves (loops A & B) are opened by operator action. A loss-of-coolant accident signal (high drywell pressure and low vessel water level) automatically closed these MOVs.

E11-F051A and E11-F051B are E/P controlled air operated throttle valves. At a certain setpoint of heat exchanger shell pressure, these valves begin to close and will completely close before exchanger design pressure is exceeded. This circuitry is powered from a nonsafeguard source. A LOCA signal initiates closure of these air operated valves. In the event that the nonsafeguard pressure reducing circuitry on E11-F051A and E11-F051B should fail, pressure relief valves (E11-F055A and E11-F055B) would maintain pressure on the low pressure system side below limits.

The AE supplied bypass valves HV-C-51-154A and B (used by bypass E11-F051A, B) will have their power sources disconnected and the valves will be locked in the closed position.

<u>Motor-Operated Valves in Series with (Testable) Check Valves (BTP-3 par 4)</u>

E11-F015A and E11-F015B (RHR shutdown cooling injection outboard valves) are manually activated motor operated valves in series with E11-F050B (testable check air operated valves), respectively. These motor-operated valves (loop A and B) ar inhibited from opening and close automatically if primary system pressure is above setpoint. Both valves use the same valve control circuit, which requires two reactor low pressure permissives before valves can open. Removal of one pressure permissive signal will close the valves.

The remaining HPLPSI valves in this discussion are required for emergency core cooling systems operation. The recommendation of BTP-3 was followed in evaluating ECCS high pressure/low pressure interlocks on an individual case basis.

Paired Motor-Operated Valves and Air Operated Check Valves

The valves listed below are paired motor operated valves and air operated check valves, which isolate low pressure ECC systems from higher pressure primary system.

E11-F017A, B, C, D LPCI injection MOVs are interlocked to prevent opening when differential pressure across the valves exceeds the setpoint. This interlock applies to manual or automatic opening. The ΔP is indicated by a permissive alarm in the control room. The normally closed core spray inboard injection valves (E21-F005 and E21-F037) and the normally open outboard injection valves (E21-F004A and E21-F004B) are interlocked by high reactor pressure (one-out-of-two twice logic) to prevent their receiving an opening signal on automatic system initiation. The inboard and outboard valves are interlocked by limit switch to prevent both valves in each loop from being opened manually at the same time during testing.