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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

'86 MAR 18 P1:15

Before the Atomic Safety and Licensing Board

OFFICE OF SECRETARY DOCKETING & SERVICE BRANCH

In the Matter of	)	
	)	
Philadelphia Electric Company	)	Docket No. 50-352-OLA
	)	(Check Valve)
(Limerick Generating Station,	)	
Unit 1)	)	March 17, 1986

LICENSEE'S ANSWER TO CONTENTIONS PROPOSED BY INTERVENOR ROBERT L. ANTHONY

Preliminary Statement

This proceeding results from the 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] On March 13, 1986, the presiding Atomic Safety and Licensing Board ("Licensing Board" or "Board") ruled that Mr. Anthony's petition had been granted, subject to a finding that at least one contention meets the requirements of 10 C.F.R. §2.714.[2]

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

2/ Philadelphia Electric Company (Limerick Generating Station, Unit 1) (Check Valve), "Memorandum and Order Ruling on Robert L. Anthony's Petition for Leave to Intervene" (March 13, 1986). The Board advised the parties by telephone of its decision.

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Mr. Anthony filed a list of proposed contentions on February 15, 1986.[3/] In a Memorandum and Order entered March 6, 1986, the Board directed Licensee and the NRC Staff to file responses to Mr. Anthony's proposed contentions by noon, March 17, 1986 unless otherwise advised.[4/] For the reasons discussed below, Philadelphia Electric Company ("Licensee") opposes the contentions 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

Before addressing each of the proposed contentions individually, Licensee will make some initial observations of general applicability regarding the admissibility of contentions. First, it is noted that this proceeding presents only very narrow questions relating to the validity of Amendment No. 1 to the Limerick operating license. Under the governing notice of opportunity for a hearing and the

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3/ See Intervenor R.L. Anthony/FOE Request for a Hearing and Petition for Leave to Intervene in the Light of the Issuance on 2/6/86 of Amendment 1 to Lic. No. NPF-39 Without a Hearing, and Petition for Stay of Operation After 2/19/86 Limit for Tests (February 15, 1986) ("Anthony Contentions").

4/ Limerick, supra, "Memorandum and Order on Licensee's Motion to Defer Answers to Petitioner's Contentions" (March 6, 1986) (slip op. at 2).

delegation of authority to this Board,[5/] this proceeding cannot be a basis for litigating safety and environmental issues which were or could have been litigated in the operating license proceeding for Limerick.[6/]

5/ See Philadelphia Electric Company (Docket No. 50-352-OLA), "Establishment of Atomic Safety and Licensing Board" (February 12, 1986). The Order stated:

> This Board is being established pursuant to a notice published by the Commission on December 26, 1985 in the Federal Register (50 F.R. 52874) entitled, "Consideration of Issuance of Amendment to Facility Operating License and Proposed No Significant Hazards Consideration Determination and Opportunity for Hearing." The amendment would revise the Technical Specifications to allow a one-time-only extension of time to satisfy a limited number of testing requirements for the excess flow check valves in certain instrumentation lines which must be performed every 18 months and which require plant shutdown.

6/ As the Appeal Board stated in the Catawba proceeding:

> Adjudicatory boards do not have plenary subject matter jurisdiction in Commission proceedings. Under the Atomic Energy Act, the Nuclear Regulatory Commission is empowered to administer the licensing provisions of the Act and use licensing boards "to conduct such hearings as the Commission may direct." The boards, therefore, are delegates of the Commission and, as such, they may exercise authority only over those matters that the Commission commits to them. The various hearing notices are the means by which the Commission identifies the subject

(Footnote Continued)

This point was driven home cogently by the Appeal Board in a license amendment proceeding for the Point Beach facility.[7/] The amendment permitted the licensee to repair degraded steam generator tubes by sleeving them rather than plugging the tubes and removing them from service. The intervenor proposed a contention concerning the effects of steam generator tube failures during accident and normal operating conditions. The Appeal Board affirmed the Licensing Board's denial of the contention and its ruling that "absent a showing that sleeving would lead to tube failures, the issue of the consequences of steam generator tube failure was not relevant to [the] amendment proceeding."[8/] The Appeal Board then stated as follows:

> In a license amendment proceeding, a licensing board has only limited jurisdiction. The board may admit a party's issues for hearing only insofar as those issues are within the scope of matters outlined in the Commission's notice of hearing on the licensing action. Here,

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(Footnote Continued)

> matters of the hearings and delegates to the boards the authority to conduct proceedings.

Duke Power Company (Catawba Nuclear Station, Units 1 and 2), ALAB-825, 22 NRC 785, 790 (1985) (footnotes omitted). See also Portland General Electric Company (Trojan Nuclear Plant), ALAB-534, 9 NRC 287, 289-90 n.6 (1979); Public Service Company of Indiana, Inc. (Marble Hill Nuclear Generating Station, Units 1 and 2), ALAB-316, 3 NRC 167, 170-71 (1976).

7/ Wisconsin Electric Power Company (Point Beach Nuclear Plant, Units 1 and 2), ALAB-739, 18 NRC 335 (1983).

8/ Id. at 339 (emphasis added).

> the notice of hearing stated [that] the proceeding would concern the repair of steam generator tubes by sleeving and the operation of the Point Beach plant with sleeved tubes. Thus [intervenor] had to put forth a cognizable claim that some element of the sleeving process gives rise to an enhanced likelihood of tube rupture and the allegedly concomitant consequences. . . . [Intervenor] was aware it had to make this showing . . ., yet it failed to provide any link demonstrating that sleeving may lead, or be related, to tube failures.9/

Similarly, in a proceeding regarding enlargement of the spent fuel pool storage capacity for the Zion plant, the Appeal Board ruled that the Licensing Board had correctly excluded testimony as to alleged inadequacies in the emergency plan entirely unrelated to expansion of the spent fuel pool.10/ The Appeal Board stated that the Licensing Board "was not empowered to reconsider whether the Zion facility should have been licensed to operate in the first instance, or whether the emergency plan approved in conjunction with that license was generally in need of revision."11/ For the same reasons, the Board in this proceeding should deny any proposed contention which seeks, in essence, to reopen the operating license proceeding to litigate safety issues such as reactor design and accident analysis for Limerick. Those

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9/ Id. (citations omitted) (footnotes omitted).

10/ Commonwealth Edison Company (Zion Station, Units 1 and 2), ALAB-616, 12 NRC 419 (1980).

11/ Id. at 426.

matters are beyond the scope of the notice of opportunity for hearing.[12/]

Second, the record in this proceeding, given the narrow issues involved, is far smaller than in an operating license proceeding. Basically, it consists of the application for the amendment, the NRC Staff's written safety evaluation and the amendment itself. Because the record is not at all lengthy and there was ample advance notice and time for Mr. Anthony to prepare contentions, the Board should insist upon strict compliance with the requirements for specificity and bases under 10 C.F.R. §2.714(b).

In Zion, the Appeal Board held that "there is no duty placed upon a licensing board by the Administrative Procedure Act, or by [the Atomic Energy] Act and the regulations promulgated thereunder, to recast contentions offered by one of the litigants for the purpose of making those contentions

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12/ In its Memorandum and Order Ruling on Robert L. Anthony's Petition for Leave to Intervene (slip op. at 9 n.5) the Licensing Board characterized the Notice of Opportunity for Hearing as inviting intervenors to seek to litigate "the ability of the instrument lines to function during the extension of time for testing." Licensee disagrees. The hearing is limited to issues directly related to the excess flow check valves as a result of the incremental 14-week period. The design of the instrumentation lines was approved by the Commission prior to the issuance of an operating license for Limerick Unit 1. The necessity for operation of the excess flow check valves assumes instrumentation line failure.

acceptable."[13/] Thus, the Board should not engage in a rewriting or reformulation of any insufficient contention. This is particularly true here because Mr. Anthony has participated in NRC proceedings over the past four years[14/] and, as a seasoned intervenor, is fully knowledgeable as to pleading requirements.[15/]

Third, the contentions proposed by Mr. Anthony are so completely lacking in technical specificity that it is impossible to understand their substance or determine the regulation or other requirement with which Licensee allegedly has failed to comply. Cursory reference to a smattering of technical reports with no apparent nexus to Amendment No. 1 does not overcome this deficiency. In short, the proposed contentions constitute only "oblique reference[s]" to

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13/ Zion, supra, ALAB-226, 8 AEC 381, 406 (1974). The Appeal Board ruled more recently that substantial reframing of contentions "would be tantamount to the raising of a new issue sua sponte - action that is now subject to immediate Commission oversight and that can be invoked only by observing special procedures. Cleveland Electric Illuminating Company (Perry Nuclear Power Plant, Units 1 and 2), ALAB-675, 15 NRC 1105, 1115 (1982).

14/ See Limerick, supra, LBP-82-43A, 15 NRC 1423, 1440 (1982).

15/ Mr. Anthony's participation in NRC proceedings is discussed at page 8, infra.

possible issues[16/] and are impermissibly "conclusional . . . barren and unfocused."[17/]

In the operating license proceeding for Limerick, the Appeal Board reached the same conclusion in ruling that Mr. Anthony's "purported contentions are unfocused and contain no attempt to identify with reasonable specificity the basis of the perceived risks" alleged in the contentions.[18/] The Licensing Board has also lectured Mr. Anthony on his failure to comply with specificity and basis requirements. In denying contentions he proposed in 1984, the Board ruled:

> FOE's contentions are very poorly pleaded. . . . [I]t is worth setting out here a list of the kinds of deficiencies the contentions so blatantly exhibit. Some are vague; others appear to be based on a belief that mere speculation that something might go wrong, or not be done, is enough to generate litigation; some appear to be based on poor reading of the documents cited by FOE; many merely cite concerns expressed in various NRC documents; and none of them give any reason why the facts alleged in them demonstrate that <u>low</u> power operation would be unsafe.[19/]

16/ <u>Illinois Power Company</u> (Clinton Power Station, Unit Nos. 1 and 2), ALAB-340, 4 NRC 27, 51 (1976).

17/ <u>Offshore Power Systems</u> (Manufacturing License for Floating Nuclear Power Plants), LBP-77-48, 6 NRC 249, 250-51 (1977).

18/ <u>Limerick</u>, <u>supra</u>, ALAB-765, 19 NRC 645, 653 (1984).

19/ <u>Limerick</u>, <u>supra</u>, "Memorandum and Order Rejecting Late-Filed Contentions from FOE and AWPP, Denying AWPP'S Second Request for Reconsideration of Asbestos Contention, Denying AWPP'S Motion to Add a PVC

(Footnote Continued)

Thus, it is not enough for Mr. Anthony merely to speculate that some event might occur or that failure of a system or component is hypothetically possible. The Commission has ruled that contentions "lack the requisite specificity" if they "do not identify any particular structures, systems or components for which it is claimed that [the licensee's activities were] not commensurate with their safety function."[20/] Or, as the board in Shoreham stated, a valid contention must "specify the particular features" of the regulatory requirement at issue and "show the nexus of those features" to safety or environmental issues directly

(Footnote Continued)
Contention and Commenting on an Invalid Inference in Del-Aware's May 17, 1984 Filing" (August 24, 1984) (slip op. at 5) (copy attached). As with the amendment in the instant case, Mr. Anthony attempted to use the low-power application in the operating license proceeding as an excuse to relitigate safety issues. Id. at 6-13. The Board sharply chastized Mr. Anthony for his total failure to meet contention pleading requirements:

> It should be clear by now how utterly indistinguishable FOE's contentions are, in their baselessness and their carelessness, from contentions hastily thrown together in an effort to achieve mere delay in the conclusion of the low power issues part of this proceeding.

Id. at 14.

20/ Pacific Gas and Electric Company (Diablo Canyon Nuclear Power Plant, Units 1 and 2), CLI-84-14, 20 NRC 285, 286 (1984).

traceable to the amendment.[21/] General averments concerning the Limerick plant, especially systems unaffected by the amendment, are insufficient.

With these basic principles as background, Licensee now addresses each of the proposed contentions.

Contentions 1 and 2. These contentions challenge the categorical exclusion of Amendment No. 1 from requirements under 10 C.F.R. Part 51 for preparing an environmental impact statement or environmental assessment for licensing actions. In relevant part, Section 51.22 states:

> (b) Except in special circumstances, as determined by the Commission upon its own initiative or upon request of any interested person, an environmental assessment or an environmental impact statement is not required for any action within a category of actions included in the list of categorical exclusions set out in paragraph (c) of this section. Special circumstances include the circumstance where the proposed action involves unresolved conflicts concerning alternative uses of available resources within the meaning of Section 102(2)(e) of [the National Environmental Policy Act of 1969, 42 U.S.C. §4332(2)(E)].

21/ Long Island Lighting Company (Shoreham Nuclear Power Station, Unit 1), LBP-81-18, 14 NRC 71, 75 (1981). The Board further stated:

> The requirement of greater specificity is necessary to provide a fair opportunity for other parties to learn precisely what the issues are, what proof, evidence or testimony is required to meet the issues, and what the Intervenor intends to adduce for its allegations.

(Footnote Continued)

> (c) The following categories of actions are categorical exclusions:
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> . . .
>
> (9) Issuance of an amendment to a permit or license for a reactor pursuant to Part 50 of this chapter which changes a requirement with respect to installation or use of a facility component located within the restricted area, as defined in Part 20 of this chapter, or which changes <u>an inspection or a surveillance requirement</u>, provided that (i) the amendment involves no significant hazards consideration, (ii) there is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite, and (iii) there is no significant increase in individual or cumulative occupational radiation exposure. [Emphasis added.]

Mr. Anthony apparently challenges the Staff's determination in its Safety Evaluation that Amendment No. 1 "meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9)" and that "[p]ursuant to 10 CFR 51.22(b) no environmental impact statement nor environmental assessment need be prepared in connection with the issuance of this amendment."[22/] To the contrary, the exclusion regarding surveillance requirements explicitly

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(Footnote Continued)

<u>Id</u>.

22/ Safety Evaluation by the Office of Nuclear Reactor Regulation, Support Amendment No. 1 to Facility Operating License No. NPF-39, Philadelphia Electric Company, Limerick Generating Station, Unit No. 1, Docket No. 50-352 at §3.0 (February 6, 1986) ("Safety Evaluation").

covers Amendment No. 1 for the reasons stated in the application for the amendment and the Safety Evaluation. Mr. Anthony has not provided any reason why the amendment at issue does not fall within the categorical exclusion. *A fortiori*, he has not provided any reason with specificity and basis. Instead, Mr. Anthony simply repeats the language of the regulation with the unadorned assertion that the eligibility criteria have not been met. These contentions state nothing to litigate and should be denied.

Contentions 3 and 10. These contentions allege that the brief extension granted by Amendment No. 1 for surveillance testing of certain reactor instrumentation line excess flow check valves "violates the maximum time limit set by NRC for [their] safe operation"[23/] and that the "amendment violates the requirements of, and intention of the Atomic Energy Act and the regulations."[24/] In effect, these contentions assert that any extension of the time for completing surveillance testing for reactor instrumentation line excess flow check valves is unsafe because a shorter period has been set forth in the plant's Technical Specifications. This is circular logic, pure and simple. Under that reasoning, the NRC could never grant a license amendment. Contrary to Mr. Anthony's rationale, extensions of

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23/ Anthony Contentions at 1.

24/ *Id.* at 3.

surveillance times have been approved in numerous other dockets.[25/]

Contrary to these contentions, the NRC's regulations clearly recognize that an amendment to a license will be issued to an applicant if it meets "the considerations which govern the issuance of initial licenses . . . to the extent applicable and appropriate"[26/] Mr. Anthony's argument that any amendment of the Technical Specification is unsafe constitutes, in reality, an unauthorized challenge to the regulations. Such a challenge is not a permissible basis of a contention.[27/]

Moreover, the issue framed by these contentions is utterly lacking in any specificity or bases. They merely constitute Mr. Anthony's ipse dixit that any extension of time for surveillance testing is unsafe. No specific issue regarding the public health and safety has been framed. Thus, the contentions fail to state any basis for contesting the conclusion by the Staff that the 18 month surveillance

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25/ E.g., Public Service Electric & Gas Company (Salem Nuclear Generating Station, Unit 1), LBP-84-5, 19 NRC 391 (1984).

26/ 10 C.F.R. §50.92(a).

27/ See Metropolitan Edison Company (Three Mile Island Nuclear Station, Unit No. 2), ALAB-456, 7 NRC 63, 65 (1978). See also Rochester Gas and Electric Corporation (Sterling Power Project Nuclear Unit No. 1), ALAB-507, 8 NRC 551, 556 n.17 and accompanying text (1978).

interval has no particular safety significance, but was simply "selected to be consistent with the maximum anticipated interval between refueling outages."[28] Accordingly, these contentions should also be denied.

Contention 4. This contention asserts that the extended start-up program for Limerick, resulting from delay in the issuance of a full-power operating license, "reinforces rather than eliminates the need for the surveillance tests on schedule" because "[c]hanges in cooling water pressure and starts and stops of the reactor could exert more strain tha[n] continuous operation."[29]

This allegation has no basis. 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. When such a situation exists, the valve will seat, checking flow downstream in the line.[30] Mr. Anthony has failed to allege any basis for asserting that any "deterioration" or "strain" occurs during normal operation or reactor shutdown when the valve is passive. This vague contention should be denied.

28/ Safety Evaluation at §2.0.

29/ Anthony Contentions at 2.

30/ See FSAR §6.2.4.3.1.5 (a copy of all FSAR references herein is attached).

Contention 5. In this contention, Mr. Anthony asserts that alleged "hazards from the malfunctioning of the valves is highlighted" by portions of the Safety Evaluation.31/ The cited portion of the Safety Evaluation states:

> Testing of the valves to verify that they check flow involves opening of the instrumentation line downstream of the valve with the reactor coolant system cold and pressurized and verifying that the valves check flow. This operation cannot be performed during normal power operation for the following reasons: (1) the performance of the test with the reactor coolant system hot, pressurized and at power would involve potential hazards to testing personnel upon opening of the line in the unlikely event that one of the valves fails to check and releases fluid that is both at a high temperature and radioactive, and (2) the opening of the instrumentation line, since the line may serve an instrumentation manifold with multiple transmitters, would result in multiple engineered safety feature system and/or reactor protection system actuations which would either constitute conditions prohibited by Technical Specifications or result in a shutdown of the reactor.32/

Contrary to Mr. Anthony's allegation, surveillance testing of these valves, as described in the Safety Evaluation, does not reflect any "hazards from the malfunctioning of the valves." The Safety Evaluation simply states that, inasmuch as testing simulates a break in the line, the

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31/ Anthony Contentions at 2.

32/ Safety Evaluation at §2.0.

person performing the test during normal power operation could be injured or contaminated in the unlikely event that a valve should fail. Further, depressurization of an instrumentation line during a test at normal power operation could initiate reactor scram and actuation of the emergency core cooling system ("ECCS").[33] These considerations in not performing testing during normal power operation do not reflect any alleged "hazards" from a possible "malfunction" of the valves during normal operation. This contention has no litigable basis and should be denied.

Contentions 6, 7, 8 and 9. In one fashion or another, each of these contentions seeks to raise as an issue some aspect of reactor design or accident analysis completely distinct from surveillance testing of check valves. Such matters may not be litigated in this license amendment proceeding because it pertains *only* to a one-time 14-week extension of the allowable interval for certain surveillance testing. If Mr. Anthony desired to litigate broader safety issues related to excess flow check valves or instrumentation lines, he should have done so in the operating license proceeding.[34] His proposed contentions in this group therefore exceed the scope of the proceeding and should be

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33/ *Id*. *See also* Application for Amendment of Facility Operating License NPF-39 at 1-2 (December 18, 1985) ("Application for Amendment No. 1").

34/ *See* note 12 *supra*.

denied. Further, they should be denied because they lack adequate specificity and bases for litigation.

Contention 6 alleges that a number of the valves in question operate in key plant systems and the "failure of one or more of these valves and instrument lines could cause radioactive releases and precipitate other failures, resulting in catastrophe."[35/] The design basis for instrumentation lines and excess flow check valves, utilizing Regulatory Guide 1.11, was previously analyzed in FSAR §6.2.4.3.1.5. The consequences of valve and instrumentation line failure were analyzed in FSAR §15.6.2. The list of excess flow check valves was contained in the Technical Specification issued originally with the low-power license in October 1984. With regard to a possible instrumentation line break and failure of an excess flow check valve, the Limerick design is based on conservative assumptions considered to be acceptable to the NRC for keeping below offsite dose guidelines under 10 C.F.R. Part 100.[36/] Mr. Anthony has not shown how a hypothetical instrumentation line or valve failure relates to the extension of time to conduct surveillance testing. Moreover, even if permitted to challenge

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35/ Anthony Contentions at 2.

36/ See FSAR §15.6.2.5. The calculated exposure at the Exclusion Area Boundary and the Low Population Zone are presented in FSAR Table 15.6-7. These values are orders of magnitude below Part 100 guidelines. See Application for Amendment No. 1 at 5.

safety analyses completed in licensing Limerick, he has not asserted anything related to the amendment which would justify reexamining these analyses.

Contention 7 asserts that failure of excess flow check valves "could precipitate other faults and even cut off the functions of instruments needed for safe shutdown."37/ This assertion is completely without basis. Nowhere does he provide any mechanism by which failure of an excess flow check valve to check flow when there has been a line break would jeopardize the capacity for safe shutdown.38/ Mr. Anthony makes no showing to the contrary.

As stated in the Safety Evaluation, the purpose of excess flow check valves on instrumentation lines is to check flow in the line when subjected to an excess differential pressure created by a line break.39/ Hence, the only consequence of a check valve failure following an instrumentation line rupture is the release of primary coolant into secondary containment.40/ Mr. Anthony has not alleged any

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37/ Anthony Contentions at 2.

38/ To the contrary see FSAR §15.6.2.3.

39/ Safety Evaluation at §2.0.

40/ See FSAR §15.6.2.4.1. As stated in the FSAR and restated in the Safety Evaluation, instrumentation lines are small diameter pipes with flow restricting orifices to reduce loss of coolant, which would be contained in the secondary containment and processed by the standby gas treatment system. FSAR §§15.6.2.3 and 15.6.2.4; Safety Evaluation at §2.0.

basis for asserting that the loss of an excess flow check valve in conjunction with rupture of the associated instrumentation line would have any consequence to safety systems or would preclude safe shutdown of the reactor. In any event, his allegations are totally irrelevant to the surveillance testing amendment.

Contention 8 cites excerpts from two studies regarding the hazards of an "interfacing systems LOCA."[41/] Mr. Anthony has demonstrated no basis for applying the conclusions of these studies to excess flow check valves at Limerick. An interfacing systems LOCA as discussed in those studies concerns the potential for failure of normally closed valves between the high pressure reactor coolant

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41/ Anthony Contentions at 2. See R. Fitzpatrick, L. Arrieta, T. Teichmann and P. Davis, "Probabilistic Risk Assessment (PRA) Insights" NUREG/CR-4405 (December 1985); Sarah M. Davis, "Insights Gained from Probabilistic Risk Assessments" (September 20, 1984) (copy of excerpts attached). Preliminarily, Licensee notes that the excerpt cited from one of the studies shows that an interfacing systems LOCA is a substantial contributor to risk at pressure water reactors such as Millstone and Seabrook. See "Probabilistic Risk Assessment (PRA) Insights" at xiii. In fact, the same study shows that for Shoreham, like Limerick, a boiling water reactor with a General Electric Mark II containment, LOCA's of any kind are an "insignificant" contributor to risk. Id. at xiv. The other study cited by Mr. Anthony shows the same conclusion for Limerick. See "Insights Gained from Probabilistic Risk Assessments" at 10 (Table 1.2). Thus, Mr. Anthony's citations do not provide any basis as regards Limerick.

boundary and connected low pressure systems.[42/] This subject is irrelevant to excess flow check valves because they are normally open and there is no associated low pressure system.[43/] No basis exists for Mr. Anthony's contrary assertion. Nor does probabilistic risk analysis of interfacing systems LOCA's at other plants have any relevance whatsoever to Amendment No. 1.[44/]

Contention 9 asserts that "instrumentation lines were found to be vulnerable to jet impingement loads from the rupture or whipping of adjacent pipes."[45/] Here again, Mr. Anthony is attempting to litigate reactor design issues which were resolved in granting Limerick an operating

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42/ Thus, as stated in the studies relied upon by Mr. Anthony, the valves of concern are located "at the high/low pressure boundary." "Insights Gained from Probabilistic Risk Assessments" at 24.

43/ Moreover, an instrumentation line break at Limerick would not result in a LOCA in the sense discussed in the two studies cited by Mr. Anthony. Loss of an instrumentation line would not affect make-up capacity for Limerick. See FSAR §15.6.2.3.1.

44/ Mr. Anthony states that certain valves of interest regarding interfacing LOCA's are "check valves in the RHR or Low Pressure Injection Lines." Anthony Contentions at 2. However, the excess flow check valves covered by Amendment No. 1 are on instrument lines outside primary containment, not Residual Heat Removal or Low Pressure Core Injection lines. See note 48, infra and accompanying text.

45/ Anthony Contentions at 2.

license.[46/] Further, Mr. Anthony is referring to an analysis regarding jet impingement of instrumentation lines inside primary containment due to the rupture of core spray piping.[47/] Excess flow check valves, however, are located outside primary containment.[48/] This contention has neither basis nor relevance to Amendment No. 1.

In short, Contentions 6, 7, 8 and 9 wholly lack any basis for challenging the safety analysis performed by the Staff in granting Amendment No. 1. Each of those contentions impermissibly attempts to reopen the operating license proceeding and would exceed the scope of this proceeding, notwithstanding the Appeal Board's teachings in Point Beach and Zion.

Contention 11. The final contention asserts that Licensee "applied [for the amendment] to cut corners on these tests."[49/] Mr. Anthony cites a period in December 1985 during which he asserts that Licensee could have shut

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46/ See Limerick Safety Evaluation Report, NUREG-0991 at 17-5 (Supp. 4) (May 1985) (copy of excerpt attached).

47/ See Torrey Pines Technology, Independent Design Review of Limerick Generating Station Unit No. 1 Core Spray System, Executive Summary, Volume 1, at 12 and Program Results, Volume 2, at 53-54 (November 1984) (referring to line breaks postulated to occur between the reactor vessel and the first isolation valve) (copy of excerpts attached).

48/ See FSAR §6.2.4.3.1.5; Application for Amendment No. 1 at 5.

49/ Anthony Contentions at 3.

the plant down "for a long enough period to carry out the check valve tests."[50/] For the reasons discussed above, the surveillance testing at issue should not be performed even during low-power operation. Moreover, the tests require about two weeks to perform,[51/] not the few days suggested by Mr. Anthony. Thus, this contention lacks any basis. Moreover, 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. This contention should be denied.

Conclusion

For the reasons discussed above, each of the contentions proposed by Mr. Anthony lacks 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

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50/ Id.

51/ See Safety Evaluation at §2.0.

failed to plead a single admissible contention, his petition should be dismissed.[52/]

Respectfully submitted,

CONNER & WETTERHAHN, P.C.

Troy B. Conner, Jr.

Troy B. Conner, Jr.
Mark J. Wetterhahn
Robert M. Rader

Counsel for Licensee

March 17, 1986

52/ 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).

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	)	(Check Valve)
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Unit 1)	)	March 17, 1986

NOTICE OF APPEARANCE

Notice is hereby given that the undersigned attorney herewith enters an appearance on behalf of the Licensee in the captioned matter. In accordance with §2.713, 10 C.F.R. Part 2, the following information is provided:

Name	-	Mark J. Wetterhahn
Address	-	Conner & Wetterhahn, P.C. Suite 1050 1747 Pennsylvania Avenue, N.W. Washington, D.C. 20006
Telephone Number	-	202/833-3500
Admission	-	Supreme Court of the United States United States Court of Appeals District of Columbia Circuit
Name of Party	-	Philadelphia Electric Company

Mark J. Wetterhahn

Mark J. Wetterhahn

Dated at Washington, D.C.,

this 17th day of March, 1986.

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USNRC

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

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CERTIFICATE OF SERVICE

I hereby certify that copies of "Licensee's Answer to Contentions Proposed by Intervenor Robert L. Anthony" and Notice of Appearance of Mark J. Wetterhahn, dated March 17, 1986 in the captioned matter have been served upon the following by deposit in the United States mail this 17th day of March, 1986:

* Mr. Ivan W. Smith, Chairman
Atomic Safety and Licensing
Board Panel
U.S. Nuclear Regulatory
Commission
Washington, D.C. 20555

* Dr. Richard F. Cole
Atomic Safety and Licensing
Board Panel
U.S. Nuclear Regulatory
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* Mr. Gustave A. Linenberger, Jr.
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Licensing Appeal Panel
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AUG 28 1984

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING BOARD

BEFORE ADMINISTRATIVE JUDGES:

Lawrence Brenner, Chairman
Dr. Richard F. Cole
Dr. Peter A. Morris

DOCKETED USNRC

'84 AUG 27 P12:18

In the Matter of	)	Docket Nos. 50-352-OL
	)	50-353-OL
PHILADELPHIA ELECTRIC COMPANY	)	
	)	
(Limerick Generating Station, Units 1 and 2)	)	August 24, 1984

MEMORANDUM AND ORDER REJECTING LATE-FILED CONTENTIONS FROM FOE AND AWPP, DENYING AWPP'S SECOND REQUEST FOR RECONSIDERATION OF ASBESTOS CONTENTION, DENYING AWPP'S MOTION TO ADD A PVC CONTENTION AND COMMENTING ON AN INVALID INFERENCE IN DEL-AWARE'S MAY 17, 1984 FILING

1. Late-Filed Contentions from FOE and AWPP

On May 31, 1984, with the safety hearings in this proceeding concluded, and only the offsite emergency planning issues and one environmental issue left to litigate, we received from Friends of the Earth (FOE), represented by Robert L. Anthony, fifteen contentions "based on new matter" opposing the Applicant's May 9, 1984 motion for an expedited partial initial decision and issuance of a low power license. Then on July 3, 1984, with only the offsite emergency planning issues

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left to litigate, we received from Air and Water Pollution Patrol (AWPP), represented by Frank R. Romano, a new environmental contention.

Since it is now years after the Limerick proceeding began, the intervenors, to have their contentions admitted, must meet not only the bases and specificity requirements of 10 C.F.R. § 2.714(b), but also the requirements of 10 C.F.R. § 2.714(a)(1) for the admissibility of late-filed contentions. The fact that FOE's contentions are in response to the Applicant's motion for a low power license does not mean they are not late-filed, for such a motion "does not give rise to a proceeding separate and apart from a pending full-power operating license proceeding." Pacific Gas and Electric Co. (Diablo Canyon, Units 1 and 2), CLI-82-39, 16 NRC 1712, 1715 (1982). No separate proceeding is required, because 10 C.F.R. § 50.57(c), which permits such motions, "does not generally contemplate that a new evidentiary record, based on litigation of new contentions, would be compiled on the motion for fuel loading and low power testing." Pacific Gas and Electric Co. (Diablo Canyon, Units 1 and 2), CLI-81-5, 13 NRC 361, 362 (1981). The record compiled in the operating proceeding can be relied on in determining whether the motion for a low power license should be granted, for

> low power testing is a normal, necessary and expected step in the life of every nuclear plant. This is true whether such testing is planned under the authorization of a separate fuel loading and low power testing license . . . or scheduled as the first step toward operation under the authority of a full power license.

Pacific Gas and Electric (Diablo Canyon, Units 1 & 2), ALAB-728, 17 NRC 777, 794 (1983). FOE addresses the criteria on late-filed contentions and thus appears to understand that the Applicant's motion has not generated a new proceeding.

AWPP must also meet the test for reopening the record, */ for, although the whole record will not close until the offsite emergency planning contentions are litigated, AWPP's new contention is completely unrelated to emergency planning, and an appealable partial initial decision is issuing on the safety and environmental part of the record. See Long Island Lighting Co. (Shoreham, Unit 1), LBP-83-30, 17 NRC 1132, 1136-38 (1983). It is arguable that FOE also should meet the test for reopening, since, besides the offsite emergency planning contentions, only an environmental contention remained to be litigated when FOE filed its new contentions, and FOE's new contentions raise only safety issues. However, we need not decide whether to apply that test to FOE's contentions, because, as we determine below, those contentions can be rejected just for lacking bases or specificity. Also, in our view, application of the test for reopening adds little, if anything, to the application of the criteria for the admission of a late-filed

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*/ The test requires that (1) the motion to reopen be timely, (2) there be new evidence of a significant safety or environmental question, and (3) the new evidence might materially affect the outcome. See e.g., Diablo Canyon, ALAB-728, 17 NRC at 800 n.66.

contention, at least when the new contentions are unrelated to issues which were litigated. This is because the factors addressed by the criteria for reopening are necessarily considered when the factors addressed by the criteria on late-filed contentions are balanced. For example, if the record is to be reopened, the new issue must be significant (Diablo Canyon, ALAB-728, 17 NRC at 800 n.66), and if a late-filed contention is to be admitted, it must be shown that the petitioner may reasonably be expected to assist in developing a sound record (10 C.F.R. § 2.714(a)(1) (iii)); but the petitioner's assistance is worthwhile, and the record sound, only if the issues the contention raises are significant. In addition, as late-filing intervenors like to point out, the extent to which admission of a late-filed contention may delay a proceeding -- a factor which must be considered in deciding whether to admit the contention -- is properly balanced against the significance of the issues the contention raises. Shoreham, LBP-83-30, 17 NRC at 1143-44.

Given our understanding of the relation between the significance of an issue in a late-filed contention and the balancing of the five criteria for admitting it, it is appropriate for us to discuss the bases and specificity of the proposed contentions before we apply the five criteria. We discuss FOE's contentions first.

Bases & Specificity of FOE's Contentions

FOE's contentions are very poorly pleaded. We discuss each contention at least briefly, but it is worth setting out here a list of the kinds of deficiencies the contentions so blatantly exhibit. Some are vague; others appear to be based on a belief that mere speculation that something might go wrong, or not be done, is enough to generate litigation; some appear to be based on poor reading of the documents cited by FOE; many merely cite concerns expressed in various NRC documents; and none of them give any reason why the facts alleged in them demonstrate that low power operation would be unsafe.

FOE does not label its first assertion a contention, probably because it raises no safety issue, but rather questions the legal sufficiency of the Applicant's motion for a low power license. FOE asserts that 10 C.F.R. § 50.57(c), the regulation which authorizes applications for low power licenses, "provides for up to only 1% of full power," and that therefore the Applicant's motion for a license for operation up to 5% of power is not authorized. To the contrary, the regulation says, in pertinent part, "The Applicant may . . . make a motion . . . for an operating license authorizing low-power testing (operation at not more than 1 percent of full power . . .), and further operations short of full power . . ." (emphasis supplied).

FOE Contention 1 asserts that a low power license cannot be issued to the Applicant until we have reached a decision on FOE's Contentions V-3a and V-3b, which deal with the ability of plant structures to withstand nearby petroleum and natural gas pipeline explosions and fires. In the partial initial decision which we are issuing in a few days, we have ruled in the Applicant's favor on both contentions.

FOE Contention 2 asserts that no low power license can be granted until the Independent Design Review (IDR), only recently approved by the NRC, has been carried through. The contention does not provide any specificity or basis, in fact or regulation, which shows either a particular unresolved safety problem or that completion of the IDR is required for low power operation. Moreover, the contention asserts no deficiency in either the program or the Staff's review of it.

FOE Contention 3 simply cites an April 30, 1984 Notice of Violation dealing with the training, responsibilities, and management supervision of System Startup Engineers. The contention provides no basis for thinking that the Staff does not adequately understand the nature of the violation or that the Staff and the Applicant will not see to it that the causes of the violation are corrected. At no point in a proceeding, but especially not after the safety and environmental issues have been litigated, is the mere citation of a Staff inspection report finding of some deficiency sufficient basis for an admissible contention.

FOE Contention 4 simply cites a May 9, 1984 letter from the Staff to the Applicant requesting more information for the Staff's review under NUREG-0737. The request appears to raise no safety issue, and FOE says nothing to the contrary. If a mere citation to a Notice of Violation cannot be the basis of an admissible contention, the mere citation to a request for information certainly cannot be.

FOE Contention 4a quotes a May 4, 1984 Staff letter to the Applicant, which accompanied NRC Inspection Report 84-05: "The inspections of the Radiation Protection Program [RPP] found that the majority of the program, needed to support fuel load and power operation, had not been established." FOE then says that the Applicant should not be allowed to load fuel until after the Program is established. We would have thought that, given the "needed . . . " clause of the very passage FOE quotes, it was obvious there would be no fuel loading before the Program was established. FOE proffers no basis to the contrary. We add that the letter also says that during the inspection, no violations were observed, and that no reply to the letter is required. The concern the letter expresses is not that fuel load might occur before the RPP was established, but that by not leaving enough time to review the RPP, the Applicant was risking delay of fuel loading. See May 4, 1984 Staff letter, ¶ 3.

FOE Contention 5 construes a May 8, 1984 letter from the Applicant to the Staff to say that procedures meant to conform to Generic Letter

83-28, "Required Actions Based on Generic Implications of Salem ATWS Events," July 8, 1983, would not be in effect until September 1, 1984. FOE then contends that there is no assurance that the Applicant will be able to put those procedures into effect by the September date. The contention would have the burden of argument put on the wrong party: The Applicant is not required to produce proof in opposition to a mere speculation that it will not be able to meet a certain date. The initial burden is on the intervenor to come up with something more than speculation.

We note also that the contention misrepresents the scope of the procedure which the Applicant says will not be in effect until September 1, 1984. FOE leaves the impression that the Applicant is referring to all the procedures meant to conform to Generic Letter 83-28. In fact, however, the Applicant is referring only to a procedure which deals with control of vendor technical manuals. See id., at 3. We note also, for reference when we apply the criteria on admission of late-filed contentions, that although the letter FOE cites is dated May 8, 1984, the letter says that the Applicant's commitment to the September date was contained in a November 10, 1983 letter to the Staff. See the May 8, 1984 letter at 2.

FOE Contention 6 asserts quite broadly that no fuel can be loaded until "further checks of quality control in construction have been carried out" and "welds passed in error by faulty inspection" corrected.

But neither of the bases the contention cites is sufficient to make the contention admissible. One of the bases FOE cites is a sentence in NRC Inspection Report 84-17/84-05, April 25, 1984: ". . . the practice of documenting a nonconforming condition and/or authorizing rework/repairs on ASME Code items on IPRN instead of NCR was of concern to the inspector." However, the very paragraph in which this sentence occurs begins, and ends, by announcing that the issue is now closed. FOE does not say why it should be opened. Again, we note for reference later that the concern the quoted sentence reports was first expressed in a 1981 inspection report.

The other basis Contention 6 cites is a welding issue marked "open" at 3-5 in the inspection report from which we just quoted. The report says that the defects about which there is still concern are minor. _Id._, at 5. Both the Staff and the Applicant have invested much effort in bringing the issue to a satisfactory close. Moreover, this item "76-06-01," labeled the "broomstick affair" by AWPP, was fully litigated as part of AWPP's Contention VI-1. Again, for reference, we note that the issue first arose in 1976. _Id._, at 3.

Contention 7 simply cites four sections from an NRC inspection report attached to an April 18, 1984 letter from the Director of NRC's Division of Licensing to the Commissioners. The four sections have to do with welding and materials substitution. The report, however, is dated June 29, 198_3_. FOE does not say why, eleven months after the

report was issued, certain parts of it should now be subject to litigation. In relation to Contentions 6 and 7, we note that FOE sat by silently while AWPP's welding contention was litigated, and was decided in favor of the Applicant.

FOE Contention 8 asserts that all the violations listed in Enclosure 2 of the May 7, 1984 letter from the Staff to the Applicant, at 27, must be "completely rectified" before fuel loading. Enclosure 2 is Region I's Systematic Assessment of License Performance (SALP) from December 1, 1982 to November 30, 1983. A chart at page 27 of the enclosure reports that for that year-long period, there were seven violations of Severity Level IV and also seven of Severity Level V, but none for Severity Levels I - III. We note that Severity Level I is the most serious, V the least. FOE neglects to mention that the letter the enclosure accompanies says,

> Our overall assessment of your performance in the construction of the facility is that there is effective management attention and involvement, oriented toward nuclear safety in all functional areas evaluated. Your achievement of a Category I rating in five of eight functional areas indicates a determination on the part of management to achieve a high level of performance.

The Applicant achieved no less than Category II in any functional area. About Category I, the SALP says, at 6, "Reduced NRC attention may be appropriate." About Category II, it says, "NRC attention should be maintained at normal levels." Id. There is here no basis for a

contention. For reference later, we note that the SALP was available at least to the Applicant as early as February 13, 1984. Moreover, the violations the SALP counts were surely the subject of Notices of Violation available much before February 1984. If a mere citation to a Notice of Violation is not sufficient basis for an admissible contention, then neither is a mere citation to a year-end count of such notices.

FOE Contention 9 says that NRC Inspection Report 84-14/84-04, April 20, 1984, identifies several differences between the FSAR and the systems as built, several unresolved construction items, and, by reporting that some containers of nuclides were thrown in trash cans, raises the question of whether the Applicant can handle radioactive material. FOE neglects to say that no violations were reported and that the Report contains accounts of measures the Applicant has taken to cure identified shortcomings, as well as plans the NRC has to maintain watch on certain areas. See, e.g., the account of the trash can incident, id., at 10. FOE does not say why litigation would increase assurance that the matters identified in the Report will be adequately dealt with.

FOE Contention 10 merely cites an April 30, 1984 Staff letter to the Applicant as basis for the assertion that "the security program at Limerick is not adequate to allow fuel loading." However, the letter points to no shortcoming in the security program, and the NRC Inspection Report attached to the letter, No. 50-352/84-13, dated April 27, 1984

says, "Implementation of the Licensee's security program is progressing as scheduled."

FOE Contention 11 is more prophecy than allegation: "PECo has moved uranium fuel to the Limerick site without waiting for a decision by the Commission on our appeal, dated April 5, 1984 from the decision of the Appeal Board, March 30, 1984. We believe the Commission will decide in our favor . . . . " The contention then incorporates by reference all the violations and deficiencies alleged in FOE's pleadings before the Appeal Board and the Commission.

The Applicant had every right to move fuel to the site. The Appeal Board lifted its temporary stay of an issuance of the Part 70 license (see Philadelphia Electric Co. (Limerick, Units 1 and 2), ALAB-765, 19 NRC 645, 658 n.22 (March 30, 1984)), and the Commission declined to stay the Appeal Board order, finding that FOE had failed to show, inter alia, that it was likely to prevail on the merits. See Philadelphia Electric Co. (Limerick, Units 1 and 2), "Order" (April 26, 1984). We note that the Commission has since declined to review the Appeal Board's decision. See Philadelphia Electric Co. (Limerick, Units 1 and 2), "Memorandum" (June 15, 1984). Thus, the Applicant was not obliged not to receive fuel before the Commission had ruled on the merits of FOE's appeal.

Contention 12 alleges that "dangers from an explosion on the railroad have not been evaluated for the hazard to fuel being

transported from outside storage to the fuel hoistway in the plant," and that FOE was "prevented from examining witnesses on the railroad blast during [litigation of] Cont. V-3a and b." This scenario is encompassed by the bases for our March 16, 1984 order on the Part 70 license application, which found no safety concern due to postulated accidents to the new fuel. Philadelphia Electric Co. (Limerick, Units 1 and 2), LBP-84-16. 19 NRC 857, affirmed, Philadelphia Electric Co. (Limerick, Units 1 and 2), ALAB-765, 19 NRC 645 (1984). See also Philadelphia Electric Co. (Limerick, Units 1 and 2), ALAB-778, 20 NRC ___ (July 23, 1984).

FOE Contention 13 alleges that the Applicant's study of the effects of high energy line breaks (HELB), sent to the Staff on May 4, 1984, is deficient because it excludes lines which operate 2% or less of the time above 200° F. or 275 psig. See id., § 2.3. FOE contends that these lines are "most subject to rupture because of the fluctuation in heat and pressure and they could trigger other breaks . . . In addition the effects of HELB breaks on fuel handling have not been evaluated . . . ." The latter issue is impliedly dealt with in our March 16 Part 70 order, and appeals of it, as just discussed. As for the lines which are not considered in the study, FOE has not proffered any specifics or basis for thinking that the lines excluded ought to be included, other than non-expert speculation by Mr. Anthony.

Last, FOE Contention 14 alleges that the Applicant's ever-optimistic projection of fuel-load dates "suggests . . . a possible glossing over of safety issues . . . . " The mere citing of some reason the Applicant might have to gloss over safety is no basis whatsoever for an admissible contention. Rather, an intervenor must proffer evidence of some glossing over.

It should be clear by now how utterly indistinguishable FOE's contentions are, in their baselessness and their carelessness, from contentions hastily thrown together in an effort to achieve mere delay in the conclusion of the low power issues part of this proceeding.

Bases and Specificity of AWPP's Environmental Contention

We come next to AWPP's new environmental contention, less apparently frivolous than FOE's contentions, but nonetheless not admissible. In discussing AWPP's contention, we take into account an unauthorized July 25, 1984 Response AWPP made to the Staff's Response to the contention, and AWPP's August 10, 1984 reply to the Applicant's Motion to Strike the unauthorized response. The Applicant's Motion is soundly argued but the unauthorized reply may have made the contention appear even less admissible.

The contention alleges that "neither Applicant nor Staff have adequately studied whether . . . routine turbine stack, or other

releases of radioactive nuclides will result in exceeding the EPA Maximum Containment Levels (MCL) for gross alpha, radium 226 [an alpha emitter], and radium 228 [not an alpha emitter]." As basis, AWPP cites "recent findings of gross alpha approaching the MCL of 5 pico Curies [per liter of water]." AWPP is concerned that reactor releases might result in closing wells -- especially municipal wells -- within ten to fifteen miles from the plant.

Neither the contention nor the unauthorized response says how a turbine stack could release alpha-emitters or radium 228. The contention and the unauthorized Response do not even proffer a basis for thinking that these elements could reasonably be expected to be released from any point in the plant. The Applicant's Response cites contrary evidence which AWPP does not address. See id. at 7 n.13. Simple fear that the plant might regularly release alpha-emitters and radium 228 is no basis for an admissible contention. Also, the contention and AWPP's later documents say almost nothing about the wells, except that they are drinking water wells in Montgomery County. But how many they are, how close they are to the plant, whether they are municipal, and, perhaps most important, how they were studied, the contention and later documents say nothing about.

Finally, the contention misrepresents the law. AWPP neglected to give a citation for its claim that the EPA MCL on gross alpha is 5 pico Curies per liter (pCi/l). The Applicant managed, though, to find what

AWPP must have been referring to: 40 C.F.R. § 141.15. However, that section sets a limit of 5 pCi/l on the two radiums AWPP lists, not on gross alpha. Section 141.15's limit on the latter is 15 pCi/l.

AWPP needn't have entered its whole direct case in order to get the contention admitted, but one would have thought that after three filings the contention might have become accurate on the law and less mysterious about the sources of the elements in question, the nature of the wells, the mechanism by which they might become polluted, and the kind of analyses Ambler performed. As is true of each of FOE's contentions, AWPP's is not pleaded with a care proportioned to the significance the intervenor attributes to the issue raised by the contention.

Application of the Criteria for Admission of Late-Filed Contentions to FOE's and AWPP's Contentions

FOE's and AWPP's contentions could be rejected simply because they lack bases, but neither do they survive application of the five factors 10 C.F.R. § 2.714(a)(1) requires to be balanced in determining whether a late-filed contention is admissible.

Two of the factors, 10 C.F.R. § 2.714(a)(1)(ii) and (iv), weigh in the Intervenors' favor: With some exceptions, no other party to the proceeding has litigated similar contentions, and the Intervenors have

no other means by which to protect their alleged interests, or at least no other means comparable to the means litigation provides.

A third factor, whether an intervenor has good cause for failure to file on time (Section 2.714(a)(1)(i)), weighs in FOE's favor on all its contentions except 5 through 8, which, as we noted when we discussed their bases, rely on material available well before April 1984 -- 1976 in one case. But AWPP has not made a case that it has good cause to file its environmental contention late, for, despite having made three filings on the contention, and despite requests by the other parties for clarification, AWPP still has not said how "recent" the "recent findings" on which AWPP bases the contention are. "Recent" can easily mean "a few years ago."

The remaining factors, Section 2.714(a)(1)(iii) and (v), weigh heavily against both Intervenors. Admission of the contentions would clearly broaden and delay the proceeding considerably, for the record is closed, and a Partial Initial Decision about to issue, on all phases of the proceeding except offsite emergency planning issues. Further, as we noted in our discussion of the bases and specificity of the contention, AWPP says very little about where the releases it is concerned about would come from, what quantities would be released and transported, how they would cause damage, or to precisely what water supplies they would cause damage. The contention is far too vaguely drafted to permit a conclusion that it does not harbor a host of issues the litigation of

which would considerably delay the proceeding. AWPP says that any finding which indicates something which can cause "over-riding economic and/or health problems, which neither Applicant nor Staff considered, must be litigated irrespective of when found." AWPP's August 10 Reply to Applicant's Motion to Strike. But AWPP has not taken the trouble to show us that these "recent findings" it refers to indicate a threat. Thus, as far as we can tell, the considerable delay we would risk by admitting this contention would be for naught.

As to the last of the five criteria, Section 2.714(a)(1)(iii), neither FOE nor AWPP has shown that its participation could reasonably be expected to assist in the development of a sound record. "When a petitioner addresses this criterion it should set out with as much particularity as possible the precise issues it plans to cover, identify its prospective witnesses and summarize their proposed testimony." Mississippi Power and Light Company, et al., (Grand Gulf Nuclear Station, Units 1 and 2), ALAB-704, 16 NRC 1725, 1730 (1982).

It is arguable that FOE has set out the precise issues it plans to litigate, but it has said nothing about witnesses. It apparently plans to rely on cross-examination. Given FOE's contentions, and past participation in the evidentiary hearing in this case, we do not foresee that cross-examination will bring about any sound addition to the record. Most of the contentions rely on documents which appear to show that the Applicant and the Staff have the matters discussed in them well

in hand. FOE offers only poor reading and suspicion to the contrary. Neither poor reading nor suspicion assists in the development of a sound record.

AWPP's showing under the Grand Gulf standard is a sort of complement to FOE's. AWPP identifies its prospective witnesses and might be said to have summarized proposed testimony, but it is nearly as vague as possible about the precise issues it plans to cover. Moreover, its summary of proposed testimony is no more than an indication of the conclusions AWPP would like drawn from the testimony. However, a proper summary, besides stating a conclusion, also summarizes evidence and argument.

On balance, the Intervenors have no one else, and no place else, to plead their cases, but they also have almost no cases to plead. Given this balance, it is reasonable to expect that admission of these contentions would only generate useless delay.

2. Denial of AWPP's Second Request for Reconsideration of Asbestos Contention and Motion to Add a PVC Contention

On June 8, 1984, AWPP moved the Board to reconsider our then three month old denial of its late-filed contention that operation of the Limerick cooling towers will present a health hazard by contaminating the air and drinking water withdrawn from the Schuylkill River with

asbestos fibers from the asbestos cement board which is used in the cooling tower drift eliminators. The Board previously had heard extensive oral argument on March 8, 1984, in order to give AWPP every opportunity to explain its position even though its papers were manifestly insufficient to support admission of the late-filed contention, even if it had been timely filed. After the argument, the Board orally denied the contention as to both the alleged hazards to the air and drinking water. We denied the contention because it lacked bases, because it was very late by over two years without good cause, and because a balancing of all the factors applicable to late-filed contentions weighed heavily against admission of the contention. Tr. 8356-60. This ruling was confirmed in our "Order Confirming Miscellaneous Oral Record Rulings" (unpublished), slip op. at 3-4 (March 15, 1984). As we had noted in our oral rulings, among many other fatal flaws in AWPP's proposed late contention, it is not sufficient basis for AWPP to say asbestos can be a health hazard, there is asbestos in the cooling towers, ergo operation of the towers creates a health hazard from asbestos. Tr. 8356-57.

After our above ruling, AWPP filed a totally insufficient motion for reconsideration, dated March 19, 1984. We summarily denied this motion on March 27, 1984.

Almost three months later, on June 8, 1984, AWPP filed a totally insufficient second motion for reconsideration. It is denied summarily.

It is too late to merit any consideration. The time limit for motions for reconsideration has been clearly and repeatedly set forth by this Board as ten days, in addition to five days for regular mail service of the ruling which is the subject of the motion. In addition, AWPP's June 8, 1984 motion fails to address, let alone successfully rebut, the many reasons set forth for our March 5, 1984 rejection of the contention.

Most recently, in what has become a frivolous fusillade of foundationless filings by AWPP on this subject, AWPP moved, on August 16, 1984, for the admission of a late-filed contention that the use of polyvinyl chloride (PVC) instead of the originally proposed asbestos splash bars in the cooling towers will contaminate the air and drinking water. On August 20, 1984, AWPP filed an "Addendum" to address the fifth criterion for admissibility of late-filed contentions. The history on this contention is that when AWPP originally filed its asbestos contention on February 15, 1984, it had alleged hazards from asbestos used in the splash bars. In fact, Applicant had altered its original plans and used PVC, and so, on March 5, 1984, AWPP modified its contention to allege hazards from the use of asbestos in the cooling tower drift eliminators, rather than the splash bars. As noted above, this modified asbestos contention was denied.

There is no good cause for this very late August 16, 1984 motion, other than AWPP's representative saying he just read something

(unidentified) about PVC causing adverse health effects. This late-filed contention is denied. It is very late even if measured from the March 1984 time when AWPP was informed directly that PVC was being used in the splash bars. Moreover, the contention lacks bases and specificity for the same reason noted by us in our March 8, 1984 ruling with respect to asbestos. It is not sufficient to allege that because PVC can produce health hazards, if it is used in the Limerick plant there will ipso facto be a health hazard caused by operation of Limerick. Also, the lack of good cause for the very late filing, the lack of apparent significance of the issue, the lengthy delay its admission after the close of the record (and just before issuance of a P.I.D.) would cause, and the lack of any evidence of a contribution by AWPP to the record on this issue (plans to contact a possible witness in the future do not suffice), weigh heavily against admission of AWPP's PVC contention.

For all the above reasons, AWPP's PVC contention is not admitted as an issue in controversy.

3. Del-Aware

It is unnecessary to recite again the many previous rulings of this Board with respect to consideration of Del-Aware's contentions, many of them late-filed, regarding the proposed Point Pleasant diversion supplemental cooling water system. Our most recent ruling is the

April 19, 1984 "Memorandum and Order Denying Del-Aware's Motions to Reopen the Record to Admit Late-Filed Contentions . . ." (unpublished).

Subsequent to this April 19 order, on May 17, 1984, Del-Aware filed additional contentions purportedly triggered by the Applicant's request for a low power operating license. In addition, Del-Aware has improperly filed a series of letters before us relating to the proposed supplemental cooling water system. In our April 19, 1984 ruling, we set forth our view that jurisdiction over Del-Aware's claims regarding the supplemental cooling water system now lies with the Appeal Board, as part of its appellate review of our March 8, 1983 Partial Initial Decision (P.I.D.). LBP-83-11, 17 NRC 413. Del-Aware now appears to recognize this, and has filed an August 3, 1984 Motion with the Appeal Board to set aside our P.I.D. This motion appears to include the matters Del-Aware previously had raised before us and the Appeal Board in its series of letters.

The only matter deserving comment at this point is the possible inference (it is far from clear) from Del-Aware's May 17 filing before us that a low power license could not be issued until it is either certain that the proposed Point Pleasant diversion found acceptable by this Board will be finally approved by State and local authorities, or that an alternative supplemental cooling system will be proposed by the Applicant and litigated before us. We disagree.

The proposed supplemental cooling water system is just that -- supplemental. It is not needed for even full power operation for certain times of the year (generally the fall through spring months when low flow and high water temperatures do not preclude use of the Limerick plant's Schuylkill River water intake). It also is not needed for safe operation of the plant, as the ultimate heat sink for safe shutdown is the onsite spray pond.

We have held several times now that unless and until the Applicant proposes an alternative supplemental cooling water system, there is no purpose in speculating whether the one found acceptable by us will not, in fact, be permitted by other authorities. See e.g., April 19, 1984 order, supra, slip op. at 9. Issuance of a low power operating license would not change this. Del-Aware provides no basis, nor does one appear, for finding that low power testing cannot be conducted at least at times (particularly from the fall of '84 into the spring of '85), if not at all times, through use of the primary Schuylkill River cooling water intake. Moreover, even beyond low power operation, Del-Aware supplies no basis, and none appears, under the Atomic Energy Act or the National Environmental Policy Act, for an illogical finding that a completed facility which meets all applicable requirements for an NRC operating license should not be permitted to operate at all, because it will not be able to operate all the time unless and until an approved cooling water system supplemental to the Schuylkill River cooling water supply is completed.

For this reason, issuance of a low power or even a full power operating license would provide no basis to alter our decision not to consider any further supplemental cooling water system issues which depend on the predictive assumption that the proposed Point Pleasant diversion will not be completed. If and when it is certain that there is a concrete different alternative being proposed by the Applicant for its supplemental cooling water needs, then and only then would the NRC have to consider the effect of any specific proposed changes on the previous assessment of environmental impacts. See our April 19, 1984 order, supra, slip op. at 9. Indeed, Del-Aware itself has argued that only the Schuylkill River, as supplemented by releases from existing reservoirs on the Schuylkill River system, should be relied on for cooling water for Limerick. If Del-Aware's proposal is in fact proposed by the Applicant and approved by the Delaware River Basin Commission (which has authority over such water allocation decisions), then there will be no supplemental cooling water system requiring a new environmental review.

IT IS SO ORDERED.

FOR THE ATOMIC SAFETY
AND LICENSING BOARD

Lawrence Brenner

Lawrence Brenner, Chairman
ADMINISTRATIVE JUDGE

Bethesda, Maryland
August 24, 1984

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

15.6 DECREASE IN REACTOR COOLANT INVENTORY

15.6.1 INADVERTENT MAIN STEAM RELIEF VALVE OPENING

This transient is discussed and analyzed in Section 15.1.4.

15.6.2 INSTRUMENT LINE PIPE BREAK

This accident involves the postulation of a small steam or liquid line pipe break inside or outside primary containment but within a controlled release structure. In order to bound the accident, it is assumed that a small instrument line, instantaneously and circumferentially, breaks at a location where it may not be able to be isolated and where immediate detection is not automatic or apparent.

Obviously, this accident is far less limiting than the postulated events in Sections 15.6.4, 15.6.5, and 15.6.6.

This postulated accident represents the envelope evaluation for small line failure inside and outside primary containment, relative to sensitivity to detection. It is summarized in Tables 15.6-1 through 15.6-7 and Figure 15.6-1.

15.6.2.1 Identification of Causes and Frequency Classification

15.6.2.1.1 Identification of Causes and Event Description

There is no specific event or circumstance identified that results in the failure of an instrument line. These lines are designed to high quality engineering codes and standards, and seismic and environmental requirements. However, for the purpose of evaluating the consequences of a small line rupture, the failure of an instrument line is assumed to occur.

A circumferential rupture of an instrument line connected to the primary coolant system is postulated to occur outside primary containment but inside secondary containment. This failure results in the release of primary system coolant to the secondary containment until the reactor is depressurized. This accident could conceivably occur also in the drywell. However, the associated effects would not be as significant as those from a failure in the secondary containment.

15.6.2.1.2 Frequency Classification

This accident is categorized as a limiting fault.

15.6.2.2 Sequence of Events and System Operation

15.6.2.2.1 Sequence of Events

The sequence of events for this accident is shown in Table 15.6-1.

The operator should isolate the affected instrument line. Depending on which line is broken, the operator should determine whether to continue plant operation until a scheduled shutdown can be made or to proceed with an immediate, orderly plant shutdown and initiate the standby gas treatment system (SGTS) or other ventilation effluent treatment systems.

As a result of increased radiation, temperature, humidity, fluid, and noise levels within the secondary containment, operator action can be initiated by any one or any combination of the following:

a. Operator comparing the readings of several instruments monitoring the same process variable, such as reactor level, jet pump flow, steam flow, and steam pressure

b. By alarm, either high or low in the control room, from the instrument served by the failed line

c. By a half-channel scram if rupture occurred on a reactor protection system instrument line

d. By a general increase in the area radiation monitor readings

e. By an increase in the ventilation process radiation monitor readings

f. By leak detection system actuation.

Upon receiving one or more of the above signals and having made an unsuccessful attempt to isolate the break, the operator should proceed to shutdown the plant in an orderly manner.

15.6.2.2.2 System Operation

Normal plant instrumentation and controls are assumed to be fully operational during the entire plant transient to ensure positive identification of the break and safe shutdown of the plant. Minimum reactor and plant protection system operations are assumed for the analysis, e.g., minimum emergency core cooling system (ECCS) flow, and pool-cooling capability. As a

consequence of the accident, the reactor is scrammed and the reactor vessel cooled and depressurized over a 5-hour period.

15.6.2.2.3 The Effect of Single Failures and Operator Errors

The initiating event is handled by a protection sequence that can accommodate additional single failures. See Section 15.9 for a more detailed discussion of this subject.

15.6.2.3 Core and System Performance

15.6.2.3.1 Qualitative Summary

Instrument line breaks, because of their small size, are substantially less limiting from a core and systems performance standpoint than the accidents examined in Sections 15.6.4, 15.6.5, and 15.6.6. Consequently, instrument line breaks are considered to be bounded specifically by the steam line break, Section 15.6.4. Details of this calculation, including those pertinent to core and system performance, are discussed in detail in Section 15.6.4.3.

Since instrument line breaks result in a slower rate of coolant loss and are bounded by the calculations referenced above, the results presented here are qualitative rather than quantitative. Since the rate of coolant loss is slow, an orderly reactor system depressurization follows reactor scram, and the primary system is cooled down and maintained without ECCS actuation. No fuel damage or core uncovering occurs as a result of this accident.

15.6.2.3.2 Quantitative Results

Instrument line breaks, because of their small size, are substantially less limiting from a core and system performance standpoint than the steam line break outside primary containment. Similarly, instrument line breaks are considered within the spectrum considered in ECCS performance calculations discussed in detail in Section 6.3.3.

Therefore, all information concerning ECCS models employed, input parameters, and detailed results for a more limiting (steam line break) event may be found in Section 6.3.

15.6.2.3.3 Consideration of Uncertainties

The approach toward conservatively analyzing this accident is discussed in detail for a more limiting case in Section 6.3.

15.6.2.4 Barrier Performance

15.6.2.4.1 General

The release of primary coolant through the orificed instrument line could result in an increase in secondary containment pressure and the potential for isolation of the normal ventilation system.

The following assumptions and conditions are the basis for the mass loss during the 5-hour reactor shutdown period of this accident:

a. Shutdown and depressurization initiated 10 minutes after break occurs

b. Normal depressurization and cooldown of RPV

c. Line contains a 1/4-inch diameter flow restricting orifice inside the drywell

d. Moody critical blowdown flow model (Ref 15.6-1) is applicable, and flow is critical at the orifice.

The total integrated mass of fluid released into the secondary containment via the break during the blowdown is 25,000 pounds. Of this total, 6000 pounds flash to steam. Release of this mass of coolant results in a secondary containment pressure that is well below the design pressure.

15.6.2.5 Radiological Consequences

Two separate radiological analyses are provided for this accident:

a. The first is based on conservative assumptions considered to be acceptable to the NRC for the purpose of determining adequacy of the plant design to meet 10 CFR, Part 100. This analysis is referred to as the "design basis analysis."

b. The second is based on assumptions considered to provide a realistic conservative estimate of the radiological consequences. This analysis is referred to as the "realistic analysis."

A schematic of the release path is shown in Figure 15.6-1.

15.6.2.5.1 Design Basis Analysis

The design basis analysis is based on NRC Standard Review Plan 15.6.2 and NRC Regulatory Guide 1.5. The specific models, assumptions, and the program used for computer evaluation are described in Section 15.10. Specific values of parameters used in the evaluation are presented in Table 15.6-2.

The assumptions and calculation methodology used are as follows:

a. Spiking factor

The activity released from the fuel to the coolant as a consequence of reactor scram and vessel depression was based on measurements during plant shutdowns (Ref 15.6-2). It was shown that for a 95 percentile probability, a total of 7 Ci of I-131 is released to the coolant for every 1 μCi/sec of prespike I-131 release. This conservative ratio was applied for all the iodine isotopes for the dose analysis. The prespike iodine releases were those that correspond to a 0.35 Ci/sec noble gases release, a design basis accident assumption.

b. Iodine concentration in coolant

The total iodine released from the fuel to the coolant was assumed to take place in a span of 5 hours, resulting in continued buildup of coolant activity during that period. The coolant activity during 0-2 hours was assumed to be constant and equal to that at the end of the first hour. The coolant activity during 2 to 5 hours was assumed to be equal to that at the end of 3-1/2 hours. This is a conservative assumption, since the rate of increase in coolant activity decreases with time.

c. Partition factor

It was assumed that 100% of the activity in the coolant that flashed into steam remains airborne and that 10% of the activity carried by the coolant water into the secondary containment becomes airborne (corresponding to a conservative partition factor of 0.1).

d. Activity in secondary containment and released to the environment

The secondary containment volume was assumed to consist of one reactor enclosure, as discussed in Section 15.6.5.5.1.2. The activity airborne in the secondary containment was assumed to be uniformly mixed by the reactor enclosure recirculation system (RERS) with an air flow rate of 60,000 cfm and a 95% efficient filter. Secondary containment air is released to the environment via the SGTS at the rate of one secondary containment volume change per day. The SGTS filter has an efficiency of 99%. The SGTS draws air from the RERS exhaust. The activity airborne in the secondary containment and the activity released to the environment are presented in Tables 15.6-3 and 15.6-4, respectively.

The calculated exposure at the exclusion area boundary (EAB) and low population zone (LPZ) are presented in Table 15.6-7.

15.6.2.5.2 Realistic Analysis

The realistic analysis was based on a realistic but still conservative assessment of this accident. The specific models, assumptions, and the program used for computer evaluation are described in Ref 15.6-3. Specific values of parameters used in the evaluation are presented in Table 15.6.2. The leakage path used in these calculations is shown in Figure 15.6-1.

Assumptions and calculations for the realistic analysis were identical to that of the design basis except for the following:

a. Total iodine released from the fuel to the coolant was assumed to be 2 Ci for every 1 μCi/sec prespike release. This is an estimated ratio for a 50 percentile probability spiking release during plant shutdowns (Ref 15.6-2).

b. The 50% X/Qs were used instead of the 5% X/Qs used for the design basis analysis.

The activity airborne in the secondary containment is presented in Table 15.6-5. The activity released to the environment is presented in Table 15.6-6. The calculated exposure at the EAB and the LPZ are presented in Table 15.6-7.

15.6.3 STEAM GENERATOR TUBE FAILURE

This section is not applicable to the direct cycle BWR; this is a PWR-related event.

15.6.4 STEAM SYSTEM PIPING BREAK OUTSIDE PRIMARY CONTAINMENT

This accident involves the postulation of a large steam line pipe break outside primary containment. It is assumed that the largest steam line instantaneously and circumferentially breaks at a location downstream of the outermost isolation valve. The plant is designed to immediately detect such an occurrence, initiate isolation of the broken line, and actuate the necessary protective features. This postulated accident represents the envelope evaluation of steam line failures outside primary containment.

This accident is summarized in Tables 15.6-8 through 15.6-12 and Figure 15.6-2.

TABLE 15.6-7

INSTRUMENT LINE FAILURE: RADIOLOGICAL EFFECTS

DESIGN BASIS ANALYSIS

	WHOLE-BODY DOSE (rem)	INHALATION DOSE (rem)
Exclusion Area Boundary (731 meters - 2-hr dose)	5.89×10^{-7}	2.33×10^{-5}
Low Population Zone (2043 meters - 30-day dose)	3.37×10^{-7}	1.76×10^{-5}

REALISTIC ANALYSIS

	WHOLE-BODY DOSE (rem)	INHALATION DOSE (rem)
Exclusion Area Boundary (731 meters - 2-hr dose)	6.96×10^{-8}	2.75×10^{-6}
Low Population Zone (2043 meters - 30-day dose)	4.81×10^{-8}	2.51×10^{-6}

NUREG/CR-4405
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PROBABILISTIC RISK ASSESSMENT (PRA) INSIGHTS

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EXECUTIVE SUMMARY

This review of four probabilistic risk assessments (PRAs) with the goal of gaining insights into nuclear plant safety, nuclear plant vulnerabilities, and PRA methodologies was conducted by Brookhaven National Laboratory (BNL) under the sponsorship of the U.S. Nuclear Regulatory Commission. The four PRAs under investigation are those for Millstone 3, Seabrook, Shoreham, and Oconee 3. This effort was not intended as a vehicle for verifying the specific details and results of these PRAs, but rather -- having accepted the results of the PRAs -- for ascertaining what the results might mean on a plant-specific and/or generic basis. For two of the four PRAs, those for Millstone 3 and Shoreham, NRC-sponsored reviews had been completed and documented, and these were utilized in the effort; for the other two, the reviews had not been completed.

This review focused on identifying the dominant (leading) initiators, failure modes, plant systems, and specific components that affect the overall core melt probability and/or risk to the public. Each PRA was analyzed with respect to these items, and plant-specific insights were drawn from the results. In addition, the various elements of the methodologies employed by the four PRAs were discussed and ranked (per NUREG/CR-3852, "Insights into PRA Methodologies").

Perhaps the most important insight with respect to nuclear safety was the following, derived from the Oconee PRA:

- The core melt probability and public risk associated with the interfacing systems LOCA (event V), as demonstrated in the Oconee PRA, can be substantially reduced by appropriate selection of operating configuration and testing procedures and prohibition of testing of the interfacing valves with the reactor at power/pressure.

The following are other overall insights gained from this study. (Plant-specific insights are discussed in connection with each PRA).

- All four PRAs were carried out with numerous refinements over the WASH-1400 effort and have yielded more realistic results.

- The core melt probability due to internal events is identical (within error bounds) for three of the plants and relatively close for the fourth (Seabrook).

- With the possible exception of the low pressure service water system initiator at Oconee, none of the PRAs shows any internal events to be "outliers."

- The dominant risk sequences represent only a small fraction (typically less than 1%) of the total contribution to core melt probability (CMP) and are characterized by loss of the containment function due to direct bypass or overpressurization.

- In the two PRAs (Millstone and Seabrook) which specifically documented risk contribution by sequence; interfacing systems LOCA represents

Over 98% of the total contribution to early fatalities. Although not specifically quantified, the Shoreham PRA appears to identify large LOCA with early suppression pool failure as its leading contributor to eary fatalities.

- The leading contributors to latent fatalities would appear to be interfacing systems LOCA, large LOCA with early containment failure, station blackout greater than six hours and RCP seal LOCA.

- The Shoreham PRA insights listed in Section 3 are driven to a large extent by one major assumption within the PRA. The PRA has adopted a generic failure to scram probability from NUREG-0460 and assumes the common mode failure of the control rods to insert to be the only contributor. The PRA states that a Shoreham-specific analysis was done and that the results were on the order of 25% lower than the NUREG but were not used in the study. Had these results been used, the CMP as well as the dominant sequences, failure modes, system failures, and component failures would all be affected.

- The various plant PRAs show wide variance as to what internal accident initiators dominated the CMP. For Shoreham boiling water reactor (BWR), anticipated transient without scram (ATWS) dominated and loss of coolant accidents (LOCAs) were insignificant. For Oconee, LOCAs contributed approximately 30% of the CMP and a large LOCA contributed 1.5 times as much as a small LOCA. Even the two Westinghouse plants (Seabrook and Millstone) were considerably different from one another. The Seabrook and the Millstone PRAs both found the CMP contribution of a small LOCA greater than large LOCA, but a small LOCA contributed 11% in Seabrook and 24% in Millstone.

- The CMP and the percentage contribution from internal and external initiators are shown below for the four PRAs analyzed.

Plant	Total Core Melt Probability (CMP)	Contribution from Internal Initiators (%)	Contribution from External Initiators (%)
Millstone	5.89E-05	76.4	23.6
Seabrook	2.30E-04	80.0	20.0
Oconee	2.54E-04	21.3	78.7
Shoreham	5.50E-05	100.0	*

*The study did not consider external events.

The main insight drawn from these results is that the usual percentage breakdown of the contribution of internal versus external initiators of about 80/20 was fully reversed in the Oconee study. The Oconee results are for the modified plant; the external initiator dominance (mainly internal floods) was even more dominant in the original plant.

INSIGHTS GAINED FROM PROBABILISTIC RISK ASSESSMENTS

Sarah M. Davis

Reliability and Risk Assessment Branch
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U.S. Nuclear Regulatory Commission

September 20, 1984

TABLE 1.2

SEQUENCE CONTRIBUTION TO CORE MELT FREQUENCY
(GROUPED BY INITIATING EVENT* -
ROUNDED TO NEAREST 5%)

PLANT NAME	LOCA	TRANSIENT	ATWS	FIRE	SEISMIC	WIND OR TORNADO
SURRY 1	65	25	10			
PEACH BOTTOM 2		70	30			
SEQUOYAH 1	95	5				
OCONEE 3	70	25	5			
GRAND GULF 1	15	70	15			
CALVERT CLIFFS 2		95	5			
CRYSTAL RIVER 3	75	25				
ARKANSAS NUCLEAR ONE 1	25	70	5			
BROWNS FERRY 1		75	25			
MILLSTONE 1		95	5			
BIG ROCK POINT	55	15	5	25		
ZION (1 AND 2)	65	20	15			
INDIAN POINT 2	10	10		40	30	10
INDIAN POINT 3	65			35		
LIMERICK 1		100				

6. Small break LOCAs appear to be dominated by stuck open safety/relief valves in BWR.

7. Depending on the location of small break LOCAs (e.g., below reactor in pedestal cavity), the result may be to fail filling the sump prior to initiation of recirculation pumps due to flow path geometry inside containment, thus failing ECCS recirculation.

8. Interfacing Systems LOCA: The likelihood of this event can be substantially reduced through strategic testing of the valves at the high/low pressure boundary. For many plants, the valves of concern are the check valves in the RHR or Low Pressure Injection lines. However, from the Indian Point PRA, additional conditions have been recognized. The motor-operated isolation valves in the RHR suction line may also be vulnerable to an Interfacing Systems LOCA event. On the other hand, since much of the piping and the RHR heat exchanger are within containment, failure of the heat exchanger or piping in this area is no longer a sequence which bypasses containment but rather a LOCA within containment that depends on the availability of emergency mitigative systems. This configuration is somewhat unusual which underscores the importance of identifying plant-specific features which may render previously identified events less likely as well as verifying the existence of vulnerabilities found in othe plants.

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nominal thickness). BPC stated that: (1) the lower allowable levels were met in the core spray penetration, (2) the main feedwater analysis had also been revised, and (3) all contaiment penetrations with flued-heads were now verified as meeting the more conservative nozzle stress intensity limits. In reviewing the other flued-head penetrations, BPC used the lower allowable levels and minimum wall thickness, where possible. Where nominal wall thickness was needed to meet allowable limits, ultrasonic testing was performed to verify the as-built thickness. Since all flued-head penetrations were checked, the question of whether the main steam and feedwater penetrations are bounding is no longer relevant. During the subsequent visit to BPC on January 15, 1985, the Staff reviewed the analyses associated with these penetrations and found them acceptable.

PFR 017, classified as a finding, documents the lack of a thermal transient calculation for approximately six Class 1 valves supplied by General Electric that were installed in the core spray and residual heat removal systems. Subsequent analyses of these valves showed that significant margin remained after thermal fatigue was considered. The corrective action program addressed both specific and generic aspects of PFR 017 and is considered to be acceptable to the staff.

PFR 021, classified as a finding, concerned the ability of a one-inch instrument line to perform its safety function after it was subjected to jet impingement loads from a postulated core spray line break. The initial qualification of the line was based on a generic qualification using results of a test conducted on a similar line, however, TPT considered the generic qualification to contain errors and an unconservative extrapolation of test data. A subsequent analysis by BPC qualified the one-inch line. PECo addressed the generic aspects of this PFR along with the corrective action for PFRs 023 and 024, discussed below. The staff finds the corrective action taken in the case of PFR 021 to be acceptable.

PFRs 023 and 024, both classified as findings, identified errors and inconsistencies in the analysis that was used to demonstrate safe shutdown capability following postulated breaks in core spray lines. BPC agreed that there were specific areas in the analysis needing clarification or correction but did not agree that plant safe shutdown capability had not been demonstrated. Nevertheless, PECo proposed to take action to review and revise, as necessary, all safety evaluation calculations associated with jet impingement and to provide a description of the methodology of the analysis, including a discussion of how worst-case single failures are identified. At the meeting in Bethesda on January 10, 1985, PECo stated that the corrective action associated with this item had been completed and that no hardware changes were required but that minor changes in documentation had been incorporated. At the subsequent visit to BPC's offices on January 15, 1985, the Staff reviewed BPC's calculations. As a result of its review, the Staff concludes that the corrective action in this area is acceptable.

PFR 019, classified by TPT as invalid, and PFR 022, classified by TPT as an observation, involved design commitments with respect to design loading combinations for ASME 1, 2, and 3 piping or pipe support systems. PFR 019 involved jet impingement loadings on core spray pipe, while PFR 022 involved jet impingement loads associated with pipe supports. These PFRs were discussed with the licensee during the January 10, 1985 meeting in Bethesda and again during the Staff's

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INDEPENDENT DESIGN REVIEW OF LIMERICK GENERATING STATION UNIT NO. 1 CORE SPRAY SYSTEM

VOLUME 1
EXECUTIVE SUMMARY

PREPARED FOR

PHILADELPHIA ELECTRIC COMPANY

GA PROJECT 2524
NOVEMBER 1984

TORREY PINES TECHNOLOGY

A Division of GA Technologies Inc.

High and Moderate Energy Line Break Analysis, Feature 10

One Observation and three Findings resulted from the review of this feature. The Observation notes that an HVAC duct which was subject to jet impingement from a Core Spray line break was not identified in the analyses for the consequences of that break. Subsequent investigation revealed redundant cold air sources so that there would be adequate cooling.

One of the Findings pointed out the use of unconservative extrapolations of test data to evaluate the adequacy of instrumentation line subject to jet impingement loads. Review of subsequent conventional computer analysis by BPC for the instrument line impinged by a jet from a broken Core Spray pipe showed the adequacy of that instrument line. However, there were other instrument lines which were considered adequate based on the same unconservative extrapolation of the test data. The PECo CAP states that analyses will be made for all such instrument lines, and that if conventional computer analysis does not show the adequacy of the instrument lines, more sophisticated analyses or tests would be performed. Only as a last resort would jet impingement barriers be utilized. The CAP adequately addresses the concerns raised by the Finding.

The remaining two Findings addressed multiple errors and inconsistencies in the analyses which were performed to show that the plant could be safely shut down following postulated breaks in the Core Spray piping. An accurate assessment of the impact of these errors would have required significant review of plant systems and equipment which were not within the CSS. This would have gone far beyond the intended scope of the TPT review. Thus the impact of the errors could not be assessed within the TPT review scope. The multiplicity of the errors also suggested that other errors could exist which were not investigated. The PECo CAP for these two Findings identifies that all plant safety analyses associated with jet impingement will be reviewed to assure that a logical prescribed methodology is followed and that all errors and inconsistencies found are corrected. The prescribed methodology includes provision for more sophisticated analyses, or, as a last resort, design modifications. The CAP adequately addresses the concerns raised by the Findings.

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INDEPENDENT DESIGN REVIEW OF LIMERICK GENERATING STATION UNIT NO. 1 CORE SPRAY SYSTEM

VOLUME 2
PROGRAM RESULTS

PREPARED FOR

PHILADELPHIA ELECTRIC COMPANY

GA PROJECT 2524
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Fluid System Performance, Feature 9

This feature encompasses the capability of the core spray system to satisfy overall functional requirements, rather than the detailed design of specific components within the system. Accordingly, the review focused on the adequacy of the pumps and line sizes to deliver the required flow under several operating modes. System performance parameters had been specified by GE, along with the pumps, specific valves, and monitoring instrumentation. BPC developed the physical arrangement and design of the system to meet GE requirements.

Twenty-two documents were examined during the course of this review. These included design and equipment specifications, process flow and P&ID diagrams, process calculations, piping isometrics, equipment drawings, and qualifications test data. The specified GE requirements were reviewed for agreement with FSAR commitments. The core spray pump specifications and test data were reviewed for compliance with system requirements. Seven representative process calculations were reviewed along with the physical piping arrangement to determine if the following had been suitably considered: (a) pressure drop and NPSH under test and accident conditions, (b) initial flow orifice size, and (c) water hammer in a selected transient. The review performed did not reveal any deficiencies which could affect the functional performance of the system.

No PFRs were issued for this feature.

The fluid flow design of the CSS was considered adequate.

High and Moderate Energy Line Break Analyses, Feature 10

The technical review of all other features addresses the capacity of the core spray system or system components to function as required under a variety of operating conditions. This feature, however, requires consideration of

components outside the core spray system to function in the event of a major failure in the core spray system. More specifically, this feature addresses the capability of the plant to be safely shut down in the event of high energy line breaks which are postulated to occur in the Class 1 portion of the core spray piping between the reactor vessel and the first isolation valve HV-1F004A. (Piping beyond this valve is designated as moderate energy.) This feature also addresses the preclusion of flooding in the core spray pump room as a result of moderate energy line breaks.

Sixteen documents were reviewed for this feature. These documents included piping isometrics, P&I diagrams, analyses and calculations, memoranda, and a test report. In addition, a walkdown was performed (in conjunction with Task D) to identify the equipment which could be potential targets for the jets arising from the postulated line breaks. The affected equipment considered in the analyses was reviewed for completeness, using the walkdown results for comparison. The analytical methodology for evaluating the consequences of equipment failure on safe shutdown capability was reviewed. Application of the methodology was reviewed in conjunction with the P&IDs considering:

a. Loss of individual devices with individual worst single failures applied.

b. Loss of devices due to common cause with worst common single failure applied.

Where the safety analyses identified components that would remain functional after being impinged upon by a jet, the supplementary calculations and bases that had been used to support the functional survival position were reviewed. Finally, the overall conclusions were reviewed for consistency with the analytical results.

The moderate energy line break analysis was reviewed to determine the depth of the flood which can be expected to impose hydrostatic loads on the door to the room containing core spray pump A. The review encompassed the