Unresolved Safety Issues in Nuclear Power Plant Licensing: Reasonable Assurance of Safety or Nuclear Shell Game?

I. INTRODUCTION

The debate over investing a large percentage of the nation's energy capital in nuclear power has sharpened in recent months. This note makes no attempt to weigh the pros and cons of nuclear energy and alternative energy sources. It does attempt to analyze the industry's regulator, the Nuclear Regulatory Commission ("NRC"), and that Agency's responsibility in the licensing process. The adequacy of that process is the focus of this discussion.

The NRC enabling statute[1] showed Congress' intent to allow the Agency maximum leeway in nurturing the young industry.[2] With

1. The Energy Reorganization Act of 1974, 42 U.S.C. §§ 5801, 5811-5820, 5841-5849, 5871-5879, 5891 (1976 & Supp. I 1977 & Supp. II 1978), created the NRC and the Energy Research and Development Administration ("ERDA") [the ERDA was dissolved and its functions were transferred to the Secretary of Energy pursuant to the Department of Energy Organization Act of 1977, 42 U.S.C. §§ 7151(a), 7293 (Supp. I 1977), and assigned, at the discretion of the Secretary of Energy, to the Federal Energy Regulatory Commission pursuant to the Department of Energy Act of 1978, 42 U.S.C. § 7151a (Supp. II 1978)]. The NRC and the ERDA were assigned distinct responsibilities which previously were the concern of the Atomic Energy Commission ("AEC") as authorized by the Atomic Energy Act of 1954, 42 U.S.C. §§ 2011-2282 (1976 & Supp. I 1977 & Supp. II 1978 & Supp. III 1979). The NRC was given responsibility for licensing the nuclear industry by a transfer of the licensing functions of the AEC under the Atomic Energy Act of 1954, 42 U.S.C. §§ 2131-2140 (1976 & Supp. II 1978), pursuant to the Energy Reorganization Act of 1974, 42 U.S.C. §§ 5814(c), 5841(f), 5842 (1976). This allocation of functions was intended to alleviate the conflict in the AEC between regulator and industry provider. The dual role was thought to foster a regulatory perspective closely aligned with industry designs. PRESIDENT'S COMMISSION ON THE ACCIDENT AT THREE MILE ISLAND, REPORT OF THE OFFICE OF CHIEF COUNSEL ON THE NUCLEAR REGULATORY COMMISSION 1 (1979) [hereinafter cited as STAFF REPORT ON THE NRC]. Whether the reorganization resolved the conflict is uncertain. *Id.* at 2 (deposition of Commissioner Ahearne).

2. *See* Power Reactor Dev. Co. v. Int'l Union of Elec., Radio & Mach. Workers, 367 U.S. 396 (1961).

substantial autonomy the Agency formulated licensing procedures that, in its view, accommodated industry's needs with demands of safety. After tracing the deferential treatment the courts accorded the Agency's determinations, the note focuses on NRC efforts to deal with the burden of safety. By focusing on the accident at Three Mile Island, the note considers the efficacy of the licensing process and the adequacy of the Agency's assessment of risks.

II. THE PROBLEM

The NRC has broad authority to formulate standards for licensing nuclear facilities, and to determine when those conditions have been met.[3] The governing statute[4] demands, in general terms, that the NRC afford adequate protection of national security and public safety.[5] The NRC regulations require a "reasonable assurance" that "licensed activities can be conducted without endangering the health and safety of the public."[6] The statutory language and the language of the regulation have been deemed interchangeable.[7]

Judicial deference to the Atomic Energy Commission ("AEC"), and its successor the NRC,[8] originated as a dominant principle in

3. 42 U.S.C. §§ 2131-2140 (1976 & Supp. II 1978).

4. The statute requires that issuance will not be "inimical to the common defense and security or to the health and safety of the public." *Id.* §§ 2133(d), 2134(d).

5. This requirement is interpreted in 10 C.F.R. § 50.57(a)(6) (1980). The licensing process entails two separate steps. The issuance of a construction permit requires a finding of "reasonable assurance that the health and safety of the public will not be endangered by operation of the facility," *id.* § 50.35(c)(2), based upon an evaluation of the design submitted in the application. A similar finding, focused on the acceptability of construction, is necessary prior to the issuance of an operating license. *Id.* § 50.57(a)(3). The NRC may issue a construction permit while safety questions remain unanswered, provided they have found "reasonable assurance" that the questions will be "satisfactorily resolved" before completion of the construction, and that the facility may ultimately be operated "without undue risk to the health and safety of the public." *Id.* § 50.35(a)(4). However, changes may be made unilaterally in the plant design described in the applicant's Safety Analysis Report ("SAR"), if those changes do not involve an item that is safety-related. *Id.* § 50.59(a)(1). Ultimately, "adequate protection to health and safety on the public" must be found. 42 U.S.C. § 2232(a) (1976).

6. 10 C.F.R. § 50.57(a)(3) (1980).

7. Citizens for Safe Power, Inc. v. NRC, 524 F.2d 1291, 1298 n.12 (D.C. Cir. 1975).

8. *See generally* Porter County Chapter of the Izaak Walton League of America, Inc. v. NRC, 606 F.2d 1363 (D.C. Cir. 1979); Citizens for Safe Power, Inc. v. NRC, 524 F.2d 1291 (D.C. Cir. 1975); Carolina Envt'l Study Group v. United States, 510 F.2d 769 (D.C. Cir. 1975); Union of Concerned Scientists v. AEC, 499 F.2d 1069 (D.C. Cir. 1974); Nader v. Ray, 363 F. Supp. 946 (D.D.C. 1973).

Power Reactor Development Co. v. International Union of Electrical, Radio and Machine Workers ("*Power Reactor Co.*").[9] The Supreme Court upheld the AEC's authority to issue construction licenses before making a conclusive safety determination, noting the congressional intent in the Atomic Energy Act of 1954[10] ("AEA") to give it wide latitude in dealing with a new technology and a growing industry.[11]

The regulations promulgated pursuant to the AEA reflect the NRC's assessment of acceptable cost in light of the expected benefit and risk involved. Compliance with the regulations creates an administrative and judicial presumption of reasonable assurance that prevails unless circumstances can be demonstrated which cast doubt on the regulations in a manner uniquely relevant to the plant in question.[12] On judicial review of agency action, petitioners normally are foreclosed from raising objections to underlying standards.[13]

The presumption of adequacy that arises from compliance with adopted regulations reflects a concern for conserving agency and court resources. But the regulations are binding on intervenors, concerned essentially with their own locales, who may have been unaware of the rulemaking proceedings. The NRC staff need not even resolve an issue during a proceeding if it finds that it is common to all nuclear plants of a specific type and may be more "appropriately" resolved in generic proceedings.[14] Consideration is then reserved for generic proceedings apart from the consideration of any license application. In tandem with the ruling in *Power Reactor Co.*[15] allowing deferral of conclusive determinations on safety until the operator license hearing, the freedom to characterize a broad issue as generic can effectively defuse most intervenor objections during the construction stage.

9. 367 U.S. 396 (1961).

10. 42 U.S.C. §§ 2011-2282 (1976 & Supp. I 1977 & Supp. II 1978 & Supp. III 1979).

11. Power Reactor Dev. Co. v. Int'l Union of Elec., Radio & Mach. Workers, 367 U.S. 396, 409 (1961).

12. Citizens for Safe Power, Inc. v. NRC, 524 F.2d 1291, 1299 (D.C. Cir. 1975).

13. Nader v. NRC, 513 F.2d 1045 (D.C. Cir. 1975). Basic health and safety regulations need not be reassessed for each new plant. Citizens for Safe Power, Inc., v. NRC, 524 F.2d 1291, 1299 (D.C. Cir. 1975).

14. Minnesota v. NRC, 602 F.2d 412 (D.C. Cir. 1979).

15. Power Reactor Dev. Co. v. Int'l Union of Elec., Radio & Mach. Workers, 367 U.S. 396 (1961).

An additional difficulty in raising an issue arises when the objection alleges insufficient consideration of potentially harmful effects. Only clear evidence of actual danger shifts the burden of response to the Agency. The NRC is not required to speculate about potential problems which lie beyond the purview of present technology or current data.[16]

The deferral of conclusive safety assurances, the insulation of regulations, the isolation of generic issues, and the unquestioning acceptance of agency conclusions in areas of uncertainty create an irrebuttable presumption for an NRC finding of "reasonable assurance." Although the impropriety of the resulting tendency toward ultimate license approval was argued persuasively in *Porter County Chapter of the Izaak Walton League of America, Inc. v. NRC*,[17] it was firmly, if sympathetically, rejected.[18] Nevertheless, the danger of inevitable licensing that results from completed plant construction, compounded by the inability to air issues during licensing proceedings, is a serious one. The situation is unhealthy for decisions crucial to the public safety.

III. LICENSING PROCEDURES

The NRC developed its regulatory perspective[19] in an atmosphere comparatively free of judicial oversight. The safety strategy underlying NRC licensing, commonly referred to as "defense in depth," reflects the NRC's desire to allow the industry the freedom of affordable, innovative development while maintaining a satisfactory level of safety. "Defense in depth" consists of the following: (1) building and operating plants with care, relying on conservative design criteria, an approved applicant Quality Assurance Plan, sufficient NRC oversight of construction and operation, and multi-

16. Citizens for Safe Power, Inc. v. NRC, 524 F.2d 1291, 1297 (D.C. Cir. 1975).

17. 606 F.2d 1363 (D.C. Cir. 1979).

18. We do not ignore appellees' fear that the inertia generated by completion of a nuclear plant, with the massive investment it represents, will sway the licensing authority from faithfully carrying out its mandate to protect the public safety, if necessary by denying an operating license. While that contention may have practical force in some instances, a court may not transform a projected tendency to inertia into a presumption of infidelity to duty.

Id. at 1370.

19. "The underpinning for our safety assurances is our licensing process." *Oversight Hearings on Reactor Safety Study Review Before the Subcomm. on Energy and the Environment of the House Comm. on Interior and Insular Affairs*, 96th Cong., 1st Sess. 5 (1979) (statement of Chairman Hendrie) [hereinafter cited as *Safety Study Hearings*].

level review of applications within the Agency that affords an opportunity for public participation,[20] (2) anticipating errors and malfunctions and providing support components and systems necessary to cope with them;[21] and (3) constructing final defense protection against major accidents.[22] "Defense in depth" attempts to be a conservative, affirmative response to the uncertainty presented by the dearth of operating data relevant to current designs.[23]

Regardless of its avowedly conservative stance, the NRC has been subject to increasing criticism concerning safety issues over the past decade. In 1977, Congress added a specific directive to the Energy Reorganization Act of 1974[24] to address the ongoing questions of technology safety. It directed the NRC to design a specific and cohesive method for identifying and dealing with Unresolved Safety Issues ("USI") in the nuclear industry. The NRC was required to submit the plan to Congress and report annually on the progress toward its implementation.[25]

In developing its working definition of a USI, the NRC interpreted the statutory mandate to include those "issues with potentially significant public safety implications."[26] Following this interpretation, the NRC endeavored to distinguish the safety issues by

20. The applications are reviewed by the Advisory Committee on Reactor Safeguards ("ACRS"). The decisions on the application by the NRC staff are reviewed by the Atomic Safety and Licensing Board ("ASLB"). 10 C.F.R. § 2.721 (1979). Appeal from an ASLB decision may be had to the Atomic Safety and Licensing Appeal Board ("ASLAB"), 10 C.F.R. § 2.785 (1979), and, in circumstances that warrant, to the NRC. 10 C.F.R. § 2.786 (1979).

21. *I.e.*, redundant safety systems and components.

22. *E.g.*, the reactor containment shell, designed to prevent releases of radiation into the surrounding area in the event of an accident.

23. "[E]verything cannot be tested, it has to rely to some extent, on analysis and experiments on components of reactors. . . . [Assurance] rests on our inspection system and on careful oversight over operating reactors . . . ." *Safety Study Hearings*, *supra* note 19, at 12 (statement of Commissioner Gilinsky). The need for conservative systems and procedures continues in spite of four hundred years of reactor operating experience in part because "the unduly rapid push to larger sizes [of reactors] has resulted in what amounts to a generation of prototypes." STAFF REPORT ON THE NRC, *supra* note 1, at 16 (Minogue deposition).

24. "The Commission shall develop a plan providing for specification and analysis of unresolved safety issues relating to nuclear reactors and shall take such action as may be necessary to implement corrective measures with respect to such issues." 42 U.S.C. § 5850 (1976 & Supp. I 1977).

25. The first annual report, NUCLEAR REGULATORY COMMISSION, NUREG-0510, IDENTIFICATION OF UNRESOLVED SAFETY ISSUES RELATING TO NUCLEAR POWER PLANTS (1979) [hereinafter cited as NUREG-0510], was issued in January, 1979.

26. *Id.* at 10.

the degree of risk they represented. The NRC evaluated safety issues presented by its staff, Advisory Committee on Reactor Safeguards ("ACRS") analyses, and industry reports of operating mishaps.[27] Of these more than one hundred issues only seventeen were labeled USI's.[28] In classifying the issues, the Commission used multiple categorizations and the screening by fault-tree/event-tree probabilistic analysis.[29]

The staff assessed and divided the issues into four categories descending from greater to lesser risk, A through D.[30] In addition, the issues, labeled "generic tasks," were divided into eight groups by activity types,[31] which were in turn assigned a safety significance rating. The application of a risk analysis to the tasks established another four categories: I—High Risk; II—Low Risk; III—Negligible Risk; IV—Not Directly Relevant to Risk. From issues found in category A, groups 1 through 3, and category I, the staff derived the final seventeen USI's. Many generic issues are under continuing study, but are not considered serious enough to be classified as USI's.[32]

Concern for these unanswered generic safety questions arises most forcefully in the adversarial context of agency licensing hearings, yet here, where the interest in examining opposing evidence before a tribunal familiar with the technology is greatest, the chance of effective examination is severely limited. However, since

27. *See id.* at app. D (Abnormal Occurrences).

28. The method used to reduce the number of safety issues has been described as "a series of disingenuous techniques." *Hearings Before the Subcomm. on Nuclear Regulation of the Senate Comm. on Environment and Public Works*, 96th Cong., 1st Sess. 102 (1979) (statement of Robert Pollard) [hereinafter cited as Pollard Statement].

29. For an explanation of fault-tree analysis in nuclear technology, *see generally* Lewis, *The Safety of Fission Reactors*, SCIENTIFIC AM., March, 1980, at 53 [hereinafter cited as *The Safety of Fission Reactors*].

30. In considering the staff categorization of generic issues an ASLB stated that "the trouble with 'lesser' is that it is a term of comparison and is not bounded. Category B items could be only a scintilla less significant that A items, although we would expect that this is not the case." Tennessee Valley Authority, 8 N.R.C. 602, 631 (1978) (Yellow-Creek Nuclear Plant, Units 1 and 2).

31. *See* NUREG-0510, *supra* note 25, at app. B.

32. The distinction between "resolved" and "unresolved" is a troubling one. "Resolved as used in the Generic Items reports refers to the following: In some cases an item has been resolved in an administrative sense, recognizing that technical evaluation and satisfactory implementation (of the solution) are yet to be completed." Letter from M. Bender, ACRS Chairman, to Joseph M. Hendrie, NRC Chairman (November 15, 1977), *quoted in* Pollard Statement, *supra* note 28, at 103.

the Agency must find reasonable assurance for construction to proceed, issues serious enough to warrant categorization as generic safety issues are not passed without notice. The Atomic Safety and Licensing Appeal Board ("ASLAB") first detailed a procedure required for staff consideration of unresolved generic safety issues in 1977.[33] As background, the ASLAB described the comprehensive NRC safety procedure, including the preliminary safety analysis report ("PSAR"),[34] the need to comply with NRC design criteria,[35] the required analysis of the Emergency Core Cooling System ("ECCS"),[36] the PSAR review by the staff, and the issuance of a Safety Evaluation Report ("SER"). The ASLAB emphasized the care displayed in licensing procedures to support the presumption of reasonableness given to staff safety determinations.

In order to overcome this presumption of reasonableness, the intervenor must do more than raise a specific "generic issue" contention. The intervenor must establish a unique safety connection between the generic issue and the plant in the proceeding over and above its safety significance for plants in general, much like the showing required to draw into question NRC regulations and standards.[37] If the issue has been mentioned in the application and the staff has identified it as generic and reasonably assured of a timely solution, the issue is effectively removed from contention in the proceeding.

Even if an issue of safety significance has not been raised in a proceeding, the ASLAB is still required to deliberate upon whether the staff review "has been adequate"[38] to support its findings.[39] The review may be approved merely by locating "an apparent basis for the Staff's decision to allow operation to go for-

33. Gulf States Util. Co., 6 N.R.C. 760 (1977) (River Bend Station, Units 1 and 2).

34. *Id.* at 765.

35. 10 C.F.R. § 50, app. A (1979); *see also id.* at app. B, detailing the quality assurance program that is required to be described in the PSAR.

36. 10 C.F.R. § 50.46 (1979).

37. Gulf States Util. Co., 6 N.R.C. 760, 773 (1977) (River Bend Station, Units 1 and 2).

38. 10 C.F.R. § 2, app. A, at 107 (1979).

39. The ability of the ASLAB to delve into substantive matters not raised by an intervenor is limited by resources and procedural restraints. "[W]hile we may give 'appropriate consideration' to a 'serious safety, environmental, or common defense and security matter . . . that has not been raised by the parties,' we are to exercise that authority 'sparingly and only in extraordinary circumstances.' 10 C.F.R. 2.760a, 2.785(b)(2)." Virginia Elec. & Power Co., 8 N.R.C. 245, 247 (1978) (North Anna Nuclear Power Station, Units 1 and 2).

ward."[40] In any case, the staff must fully outline the safety issue and either its resolution or the staff's plans to resolve it in the SER, and furnish sufficient information to acquaint the interested public with the full implications of the situation.[41] A number of general categories allow the staff to approve an application without in-depth review of these issues in the proceeding.[42]

This approach shields safety issues from effective scrutiny, buying time while construction continues and pressure for ultimate approval builds. The staff evaluates the safety significance of a questioned component or system in terms of the function it serves and the risk it represents for the total system on the basis of design testing data. If that evaluation does not reveal an immediate solution, the staff nevertheless may approve the application including the questioned component (and the ASLAB may affirm the approval) upon the claim that future research is likely to reveal a satisfactory solution. Only hard scientific evidence, often unavailable in the comparatively young technology, can overcome a proffered "reasonable basis" for approval. In the absence of an intervenor contention, all that is needed to pass scrutiny is a "plausible" justification.[43]

At most, the NRC's decision on the River Bend Station ("*River Bend*") required that the staff assessment of generic issues with

40. Virginia Elec. & Power Co., 8 N.R.C. 245, 249 (1978) (North Anna Nuclear Power Station, Units 1 and 2). The ASLAB only considered whether the generic safety issues were considered "in a manner that is at least plausible." *Id.* at 248 n.7.

41. Gulf States Util. Co., 6 N.R.C. 750, 775 (1977) (River Bend Station, Units 1 and 2).

42. The categories include: "(a) the problem has been resolved for the reactor under study, (b) a resolution can reasonably be expected before operation, (c) there will be no safety implications until after years of operation and alternative means will exist to avoid undue risk to the public, (d) current standards are believed adequate but confirmatory studies are desirable while licensing continues, (e) a problem is so unlikely to occur as to be an incredible event, (f) the task is for the purpose of resolving unclear, conflicting, or impractical requirements of the regulations, or (g) presently adequate criteria can be improved." Tennessee Valley Authority, 8 N.R.C. 602, 627-28 (1978) (Yellow Creek Nuclear Plant, Units 1 and 2); *see also* NUREG-0510, *supra* note 25, at app. A-2. The enumeration of the categories was preceded by this caveat. "The Board has not independently evaluated the accuracy of the Staff's description of the problem, the sufficiency of the plan for resolving the problem, nor whether the basis for licensing in face of the problem is correct." Tennessee Valley Authority, 8 N.R.C. 602, 627 (1978) (Yellow Creek Nuclear Plant, Units 1 and 2).

43. This standard was reasserted in Pennsylvania Power & Light Co., 9 N.R.C. 291, 311 (1979) (Susquehanna Steam Electric Stations, Units 1 and 2).

"potentially significant public safety implications"[44] be contained in the SER. In its consideration of the Yellow Creek Nuclear Plant ("*Yellow Creek*") the NRC interpreted that decision to require only a description of the safety problem and its relation to the facility, an explanation of the program for its solution, and a rational basis for approval of licensing or continued operation.[45] Though these requirements *appeared* comprehensive, the staff managed to avoid their full impact for most generic issues by simply redefining them as not having the "potentially significant public safety implications" that triggered the River Bend requirement. The ASLAB's response to this maneuver aided the entrenchment of staff actions not already insulated from searching review by the characterization of a safety issue as "generic:"[46]

> [t]his is a conclusion of law and fact. Normally such conclusions are to be made by the adjudicatory boards based upon the evidentiary record. In this instance, however, we accept the Staff's conclusion because it is also a working conclusion which must be made by the Staff in the discharge of its responsibilities. It is within the Staff's, not the Board's, discretion to determine in the first instance which tasks require resolution before others and whether licensing may safely proceed without a program for resolution of the tasks.[47]

The ASLAB maintained a limited review by requiring that the record support the reasonable assurance that any significant defects could be resolved when necessary. However, support for this assurance rested, in part, one step further removed from the substantive issue, depending on the soundness of the staff programs for review of generic issues and their suitability for predicting when a resolution is needed, rather than on detailed data revealing the safety implications of postponing solution of a particular issue.

The treatment given to one controversial USI exemplifies both the cursory nature of the ASLAB's review, and the dangers evident in restricting the examination of an issue to a generic proceeding. Anticipated Transients Without Scram ("ATWS") describes a postu-

44. Gulf States Util. Co., 6 N.R.C. 760, 775 (1977) (River Bend Station, Units 1 and 2).

45. Tennessee Valley Authority, 8 N.R.C. 602, 625 (1978) (Yellow Creek Nuclear Plant, Units 1 and 2).

46. Minnesota v. NRC, 602 F.2d 412 (D.C. Cir. 1979).

47. Tennessee Valley Authority, 8 N.R.C. 602, 633 (1978) (Yellow Creek Nuclear Plant, Units 1 and 2).

lated "anticipated transient" (a deviation from normal operating conditions or an abnormal event) occurring without a "SCRAM" (a rapid shutdown of the nuclear reactor by dropping the control rods into the core to halt fission). In *Yellow Creek*, the ASLAB singled out the treatment of the ATWS generic issue[48] as not comporting with the requirements of *River Bend.*[49] The staff had reported that a reasonable basis existed to conclude that resolution of the issue would be reached before the Yellow Creek nuclear plant was put into operation, and had asserted that there was little chance of an ATWS causing a disaster,[50] but the staff had failed to describe fully, as required, the nature of an ATWS. However, since the proceeding was uncontested, the ASLAB simply took official notice of the definition of an ATWS[51] and concluded that the River Bend requirements were satisfied. This action indicates that the River Bend requirements may be met without using updated information. The lack of independent scrutiny or adversarial examination almost assures that even incomplete information will meet the requirement.

The dispute over ATWS as a generic issue focused on whether the improbability of its occurrence and the uncertainty of its consequences warranted the NRC's continued exclusion of its resolution from the minimum safety requisites of the design basis for nuclear power plants.[52] An NRC report issued in April, 1978,[53] concluded that the expected frequency of transients placed in doubt the reliability of current scram systems in light of asserted "safety objectives."[54] However, the staff declined to follow the report recommendations to include ATWS in the design basis, advising further study as a USI, and delaying resolution to allow fuller considera-

48. NUREG-0510, *supra* note 25, at app. A-8; THE PRESIDENT'S COMMISSION ON THE ACCIDENT AT THREE MILE ISLAND, REPORT OF THE PRESIDENT'S COMMISSION ON THE ACCIDENT AT THREE MILE ISLAND 200 (1979) [hereinafter cited as TMI REPORT]; *see also The Safety of Fission Reactors*, *supra* note 29, at 59.

49. Gulf States Util. Co., 6 N.R.C. 760 (1977) (River Bend Station, Units 1 and 2).

50. Tennessee Valley Authority, 8 N.R.C. 602, 628 (Yellow Creek Nuclear Plant, Units 1 and 2). It should be noted that the risk associated with an ATWS was deemed sufficient to warrant its inclusion in the final list of USI. NUREG-0510, *supra* note 25, at 16.

51. ATOMIC ENERGY COMMISSION, WASH-1270, ANTICIPATED TRANSIENTS WITHOUT SCRAM FOR WATER-COOLED POWER REACTORS (1973).

52. *See* NUREG-0510, *supra* note 25, at app. A-8.

53. NUCLEAR REGULATORY COMMISSION, NUREG-0460, ANTICIPATED TRANSIENTS WITHOUT SCRAM FOR LIGHT WATER REACTORS (1978).

54. NUREG-0510, *supra* note 25, at app. A-9.

tion of the cost factor involved. Thus the same issue, whose nature denies resolution in the context of an individual plant licensing and whose current implications were deemed explained in *Yellow Creek* (by referring to a five-year old report without acknowledging recent controversial information), was relegated once again to administrative limbo.

IV. THREE MILE ISLAND

On March 27, 1979, a former NRC staffmember stated to a congressional subcommittee: "[w]ithout a searching reexamination of past licensing decisions, there is a real possibility that one of the current unresolved safety issues or other undetected errors will cause a catastrophe."[55] Less than twenty-four hours later a combination of component malfunctions, poor control room design and operator errors contributed to the accident at Three Mile Island.[56]

The accident was precipitated by a transient. An interruption of feedwater to, and steam from, the power plant steam generators reduced the removal of heat from the reactor coolant system. In order to relieve the pressure build-up, and in response to it, the pilot-operated relief valve ("PORV") automatically opened. The SCRAM mechanism activated and operated smoothly. However, forty seconds into the accident, inexplicably closed feedwater block valves thwarted the backup feedwater system. Operators identified the problem and opened those valves eight minutes later. The main accident ingredient was the failure of the PORV to close after the initial reduction of pressure in the first minute and the subsequent failure of the operators to identify the malfunction and close the valve "manually." The failure to close the PORV was a result of control panel indicator ambiguity and inadequate operator training to deal with the chain of events as it unfolded. The undetected valve malfunction caused the operators to misconstrue the situation within the reactor and take actions which further diminished the effectiveness of the cooling system already impaired by the open PORV. The PORV remained open for two hours and twenty minutes, contributing to a partial melt of the reactor core.

55. Pollard Statement, *supra* note 28, at 104.

56. This note does not discuss in detail the events or causes of the accident. *See generally* TMI REPORT, *supra* note 48; *see also* 1 THE PRESIDENT'S COMMISSION ON THE ACCIDENT AT THREE MILE ISLAND, REPORTS OF THE TECHNICAL ASSESSMENT TASK FORCE 8-12 (1979) [hereinafter cited as 1 TECHNICAL TASK REPORT], for a summary of the accident.

The significance of the evaluation of the accident lies not in its identification of hardware weaknesses, but in its revelation of modes of analysis which allowed significant safety hazards to go undetected. No plant system or component can be expected to work flawlessly at all times. Indeed, the NRC does not rest its safety determinations on a guarantee of designs, but rather on its ability to anticipate dangers and avoid dire consequences through establishment of conservative design and oversight procedures, redundant safety systems, and last-line defenses; *i.e.*, by a "licensing process" that effectuates the "defense in depth" concept.[57]

In evaluating plant systems and components the NRC design basis analysis focuses on "serious event" prevention by postulating the most serious generic occurrences and planning for their prevention. The assumption in this approach is that the evaluation will cover lesser events of the same type as a consequence.[58] The loss of coolant accident ("LOCA") at Three Mile Island was a lesser event. The NRC had not considered that a minor LOCA could precipitate and be accompanied by such events, or that a lesser LOCA could perform a role different in kind from the major LOCA.[59]

The failure to recognize this disjuncture has been attributed to the NRC's use of "single failure" analysis,[60] which posits accept-

57. *Safety Study Hearings*, *supra* note 19, at 5 (statement of Chairman Hendrie).

58. STAFF REPORT ON THE NRC, *supra* note 1, at 65-69.

59. This possibility had been raised by an ACRS consultant in 1977. *Id.* at 68. However, the report was not deemed to have identified a generic safety problem and, thus, never reached the NRC. *Id.* at 50-52. Had the problem been identified as a serious generic one by the ACRS, it is uncertain what practical effect might have resulted. Although the ACRS is responsible for reviewing applications, safety hazard reports, and generic safety issues, its role is strictly advisory. *Id.* at 46-47, 54-55. The ACRS had made a recommendation that further evaluation be done on the causes and likely course of various accidents *before* commercial operation of TMI Unit 2. The staff removed the issue from the licensing process by defining it "generic." *Id.* at 44.

In addition, a PORV transient, similar to the TMI transient, had occurred in the Davis-Besse-I power plant on September 24, 1977. 4 THE PRESIDENT'S COMMISSION ON THE ACCIDENT AT THREE MILE ISLAND, REPORTS OF THE TECHNICAL ASSESSMENT TASK FORCE 197 (1979) [hereinafter cited at 4 TECHNICAL TASK FORCE]. This crucial information fell prey to the lack of comprehensive agency mechanism to examine and identify operational data for safety significance. *Id.* at 79, 114. Although the PORV had malfunctioned numerous times at various plants, the occurrences never were regarded as safety-related, but simply were listed under "current events" as "Valve Malfunctions." *Id.* at 44, 251; *see also* 1 TECHNICAL TASK FORCE, *supra* note 56, at 59.

60. 10 C.F.R. § 50, app. A (1979). "A single failure means an occurrence which results in the loss of capability of a component to perform its intended safety func-

ability of a system upon its ability to function with a theorized single, isolated component failure. Note that the possibility of associated or redundant component failure is not considered, and Three Mile Island involved a multiple-failure accident.[61] The inadequacy of this analysis was recognized on a systems level before Three Mile Island.[62] Nevertheless, the staff felt that "review procedures and acceptance criteria" were sufficent to provide reasonable assurance of safety while a study that was "expected to confirm" the belief continued. The NRC failed to recognize that the interrelation of redundant systems, whose functions have not been fully integrated in anticipation of multiple-failure sequences and have not been evaluated for safety significance in such cases, can actually have dangerous consequences, as in the Three Mile Island accident.[63]

The fact that the PORV could fail in an open position was an acceptable possibility, since manual override was available along with the backup ECCS. But the NRC does not consider failure of associated sequences in its assessment.[64] In addition, use of the "single failure" analysis allowed the staff to eliminate PORV from the category of "safety-related components and systems."[65] The "safety-related" label triggers stringent design, construction and operation requirements, but to qualify an item must be connected to a safety system and must be necessary to its operation in a "single failure" analysis.[66]

The PORV was not safety-related because it had a block valve isolating it from the primary system. The block valve was not safety-related, since the PORV isolated it. The ironic result was that, *because of the mode of analysis*, the safety precaution isolated potentially critical components from rigorous scrutiny and regulation.[67] Espousing redundancy as a significant safety precaution, and

tion." If a component, with a postulated single malfunctioning element, may perform its safety function because a redundant element compensates for its malfunctioning counterpart, it passes single failure analysis.

61. STAFF REPORT ON THE NRC, *supra* note 1, at 69-70.

62. NUREG-0510, *supra* note 25, at apps. A-12, A-13 (USI No. 9, "System Interaction in Nuclear Power Plants (A-17)").

63. 4 TECHNICAL TASK REPORT, *supra* note 59, at 17, 18, 95-99.

64. *Id.* at 206-07.

65. STAFF REPORT ON THE NRC, *supra* note 1, at 62-63.

66. Nonsafety-related items may be changed in the applicant's design plans without NRC consultation. *Id.*

67. *Id.* at 65; 4 TECHNICAL TASK REPORT, *supra* note 59, at 19.

then using it as justification for eliminating isolated items from a review that would justify reliance upon them, appears unreasonable—and has proven dangerous.

This raises a series of crucial issues. If acceptability is based upon isolated analysis without evaluation of the effect of a weakness on the overall safety of the plant, how secure can the licensing judgment be that approves a plant operation with a generic USI of uncertain proportions? Specifically, how can the ATWS issue be resolved for a particular plant without acceptable design or operating criteria *applicable to all reactor systems*, while depending on the "isolation" mode of analysis discussed above? If there is an incremental addition to the risk associated with the plant in question, how is it determined that it is acceptable, if the risk to the entire system is not analyzed and measured? Is it reasonable to find a "reasonable assurance," based upon qualitative evaluations of separate systems without an delineation of risk in quantitative terms that would inform an integrated determination of safety?

V. THE ANALYSIS OF RISK

The consideration of risk in individual determinations is vital. The difficulty in isolating and measuring risk has generally led to its subsumption in cost/benefit analysis, rather than being central to the determination of safety. To ask a decisionmaker to identify risk is not the same as asking him to justify his decision based upon it.[68]

Congress wanted to understand the risk nuclear power creates before renewing the Price-Anderson Act of 1975,[69] which limits liability in the nuclear industry. For the first time, Congress requested a documented evaluation of the safety of the nuclear plant itself, rather than relying on evaluations of the licensing procedures. It commissioned the Reactor Safety Study (WASH-1400), popularly known as the "Rasmussen Report."[70]

The report concluded that nuclear hardware operated safely; it thus delimited the risk associated with nuclear power plants. This

68. The cost/benefit decision of "safe enough," as embodied in regulatory standards, is relegated to the NRC's discretion. Citizens for Safe Power, Inc. v. NRC, 524 F.2d 1291, 1299-300 (D.C. Cir. 1975).

69. Atomic Energy Damages Act, 42 U.S.C. § 2210 (1976).

70. According to the statement of Rep. Morris Udall, it was in reliance upon the conclusion of the Rasmussen Report that the Act was hastily renewed. *Safety Study Hearings*, *supra* note 19, at 1.

led to the popular notion that the risk of dying as a result of a nuclear accident was roughly equal to the risk of being hit by a meteor. The NRC relied on the report more for public relations purposes than to support specific licensing decisions.[71]

The Lewis Report,[72] commissioned by the NRC to evaluate the Rasmussen Report, subsequently discredited the accuracy of the overall estimate of the risk associated with nuclear power plants contained in the earlier study. As a result, the NRC, without a documented foundation for its safety assessments, reverted to its reliance on licensing procedures to support its claim of overall industry safety.[73]

The Rasmussen Report, despite its weakness, did develop valuable techniques of probabilistic analysis for assessing risk in specific areas of the nuclear industry where sufficient data is available. The discrediting of the report's ultimate finding caused the valid techniques which it did contain to be all but overlooked. This was unfortunate because probabilistic analysis can be adapted to identify weak links and to assign priority to safety research.[74] The ability to perform an integrated analysis of risk[75] is a necessary tool for regulating a developing, complex and dangerous technology.[76] The acceptability of a risk may be a question for cost/benefit balancing.[77] Nevertheless, since the required finding of reasonable assur-

71. *Id.* at 9 (statement of James Weaver).

72. NUCLEAR REGULATORY COMMISSION, NUREG/CR-0400, RISK ASSESSMENT REVIEW GROUP REPORT TO THE NUCLEAR REGULATORY COMMISSION (1978).

73. *Safety Study Hearings*, *supra* note 19, at 5-10 (statement of Chairman Hendrie).

74. 1 TECHNICAL TASK REPORT, *supra* note 56, at 85-86.

75. '[R]isk' means the extent of the hazard of injury or destruction to persons or property. The term is usually expressed as the product of the probability and the magnitude of a set of consequences. Risk is not synonymous with uncertainty (which cannot be defined by a probabilistic statement, i.e. unfalsifiably), nor with the concept of cost.

Lovins, *Cost-Risk-Benefit Assessments in Energy Policy*, 45 GEO. WASH. L. REV. 911, 917 n.30 (1977).

76. "There are many places within the Commission, in the orientation of the research program, in the allocation of inspection resources, in the definition of regulatory constraints, . . . in which a foundation based on credible probabilistic calculations is more sound than one based on that elusive item known as 'engineering judgment.' " *Safety Study Hearings*, *supra* note 19, at 32 (statement of Harold Lewis, Chairman of the Risk Assessment Review Group).

77. In response to an inquiry regarding the NRC's standard of safety for the public protection, Chairman Hendrie prepared the following response:

A working definition of "adequate protection" is related to a working definition of "unacceptable risk." Both terms have been the subject of considerances demands adequate assessment of the risks inherent in making the multiple trade-offs among sophisticated systems, the method of assessment used to justify such decisions must be adequate. Without probabilistic analysis the NRC is left without an adequate methodology for risk assessment.

Since successive reports[78] questioned the assessments of data and the application of the engineering judgments underlying the assurance of safety, the NRC has a duty to produce a more reliable methodology. Congress has decided that nuclear technology is beneficial to the nation and the responsibility for implementing this policy judgment rests with the NRC. However, it must be made clear that the engineering judgments used to assure the reliability of individual components and generic systems are inadequate when applied to a complex, hybrid nuclear plant.[79]

If the dangers associated with a particular USI are clear and can be confined,[80] justifications exist for approving licenses in spite of them. The added oversight accorded to the USI by redundant support systems may actually be a conservative precaution (rather than simply a decision to postpone consideration calculated upon uncertainty), if the relevance of the USI to the integrated system is satis-

able study lately, without complete resolution. I expect the NRC will devote increased attention to these concepts, as will other branches of government and the public. At the present time, I would describe a regulated technology, such as nuclear power, as providing adequate protection of the public health and safety if each of the following criteria are satisfied:

1. It contributes only negligibly to the overall risk to public health and safety. (As used here, risk involves both the probability that an event with harmful results occurs and the consequences of the event.) That is, the risk from the regulated technology should be small compared to other risks that the public already knowingly accepts, both for "routine" events (high probability, low consequences) and for "catastrophic" accidents (low probability, high consequences).

2. The risks from the regular technology is no greater than the risks from the two basic alternatives:

(*a*) Other economically viable means for accomplishing the same purpose, in this case large scale production of electricity.

(*b*) Doing without, in this case not using large sources of electrical power.

3. Risks for future generations are no greater than those "acceptable" to the present generation under criterion 1.

Safety Study Hearings, *supra* note 19, at 16-17 (statement of Chairman Hendrie).

78. NUCLEAR REGULATORY COMMISSION, NUREG-0578, TMI-2 LESSONS LEARNED TASK FORCE: STATUS REPORT AND SHORT-TERM RECOMMENDATIONS (1979); TMI REPORT, *supra* note 48.

79. *See* note 23 *supra*.

80. The danger may not be easily circumscribed and, in fact, may exist because of deficiencies in understanding and data which are relevant to the consequences of a malfunction. *See*, *e.g.*, note 62 *supra*.

factorily examined and understood. Sound engineering judgments alone, however, are not sufficient to assure adequate integration and successful resolution of an USI in an individual nuclear facility. A credible overall systems analysis is needed to achieve that end.

The NRC has been urged to use the methodology employed by the Rasmussen Report in identifying risk bases where the data is sufficient.[81] If risks were quantified and incremental values were assigned to components and subsystems, representing the probable effects of malfunctioning on overall risk, individual safety decisions would take on a more credible air.[82] Probabilistic analysis would aid in assessing and integrating engineering judgments, though the ability to quantify would not, in itself, be a solution.

Just as computer accuracy is subject to input limitations, probabilistic analysis is prey to inadequate communication and insufficient information. If it were applied uniformly to operating data to determine component safety significance and if it supplemented a replacement of "single failure" analysis in giving priority to components and systems for "safety-relatedness," probabilistic analysis could be a significant aid.[83] Assurances of safety that include cost/benefit balances of the marginal utility of additional safety elements require a system of quantifying risk prior to the balancing. Instead of the explicit standard which probabilistic analysis would produce, a threshold assessment of "safe enough" is now offered for the total risk presented by a specific plant, and is justified by a vague comparison to alternative means of producing the same energy.[84]

VI. THE PROBLEM OF JUDICIAL REVIEW

If serious deficiencies have been revealed in the regulatory system's assessment of individual plant risk, are they insulated from effective judicial review?[85] This question is directed to the degree

81. *See* note 76 *supra*.

82. Chairman Hendrie commented that due to insufficiency of data and limitations on assessment methodology, he did not support, at that time, another major effort to demonstrate the *overall* safety of the nuclear industry through probabilistic risk-assessment. *Safety Study Hearings*, *supra* note 19, at 10 (statement of Chairman Hendrie). The suitability of the analysis for more limited purposes is not denied. Note that the TMI accident was within the bounds of the Rasmussen Report assessment. 1 TECHNICAL TASK REPORT, *supra* note 56, at 86.

83. 1 TECHNICAL TASK REPORT, *supra* note 56, at 85-86.

84. *Safety Study Hearings*, *supra* note 19 (statement of Chairman Hendrie).

85. Judicial review of any final order is authorized by 42 U.S.C. § 2239 (1976), as provided in 28 U.S.C. §§ 2341-2351 (1976 & Supp. II 1978). Exclusive jurisdiction

of deference afforded the NRC and the ensuing reluctance by the judiciary to deal with technical issues, rather than to the specifically mandated standard of review.[86] In order to provoke adequate agency consideration and response, and in order to trigger searching review of agency deliberations, intervening parties in licensing proceedings must raise substantial questions uniquely relevant to the plant involved,[87] and produce supporting evidence, to rebut the strong presumption afforded to NRC decisionmaking.

The adequacy of a finding by the NRC that a plant will be operated without undue risk to the public depends on a reasonable assessment of the risk involved. An intervenor must show affirmatively that the assessment, based on numerous, diffuse judgments and analytic systems linked and synthesized by a complex regulatory organism, is methodologically unsound. Individual components or standards can be questioned in specific rulemaking proceedings, but the dearth of operating data makes it difficult to overcome the presumption in the staff's favor. The alternative is to question the methodology used to ascertain the risk involved in plant licensing and to insure that the danger is not "undue."

Questioning the methodology is advisable where evidence can be presented to indicate untrustworthy procedures.[88] Nevertheless, presumption of an accurate methodology still lies in favor of the NRC, even though risk assessment is not peculiarly within NRC expertise. To overcome the presumption, an intervenor might need to demonstrate an effective, alternative methodology with evidence indicating appreciably larger risks that the NRC found.[89] A court

for review of final NRC orders reviewable under 42 U.S.C. § 2239 (1976) is granted to the United States courts of appeals by 28 U.S.C. § 2342(4) (1976 & Supp. II 1978).

86. Regardless of the standard of review the court still must determine "whether the decision was based on a consideration of the relevant factors and whether there has been a clear error of judgment." Citizens to Preserve Overton Park, Inc. v. Volpe, 401 U.S. 402, 416 (1971). Thus, the degree of expertise that the court assigns to the agency, and the accompanying weight it gives its judgments, may be more crucial than the particular standard of review. *See* J. JAFFE, JUDICIAL CONTROL OF ADMINISTRATIVE ACTION 576 (1965).

87. *See* text accompanying notes 37-40 *supra*.

88. For example, "single failure" analysis, as used to determine safety-related elements, disables the decisionmaker from identifying relevant factors, *i.e.*, the effect of a failure of a redundant element on the risk that the system represents to safety, or the likelihood of a common cause failure of redundant elements causing serious danger.

89. Such a burden is heavy not only because the methodology adopted for nuclear technology is young, but because the only "concrete" estimation of acceptable

might demand such a showing not only because inconclusive evidence favors the party with the presumption, but also because of the potentially unmanageable proportions of an allegation that is, in fact, an indictment of the NRC's fundamental ability to address its primary responsibility.[90] However, showing clear inadequacies in methodology should be sufficient to overcome the presumption.

The courts have wrestled with the general issue of risk in cases dealing with agency judgments involving safety determinations.[91] *Ethyl Corp. v. EPA* ("*Ethyl Corp.*"),[92] in upholding an agency determination that relied on a cautionary finding of risk from scientifically inconclusive evidence, cited *Amoco Oil Co. v. EPA*[93] approvingly: "[w]here . . . the regulations turn on choices of policy, on an assessment of risks, or on predictions dealing with matters on the frontiers of scientific knowledge, we will demand adequate reasons and explanations, but not 'findings' of the sort familiar from the world of adjudication."

It is the NRC's duty to disapprove an application where the public health and safety is unduly risked, whereas the Administrator in *Ethyl Corp.* was charged with promulgating conservative regulatory standards if he found the public was endangered. Under the NRC's mandate, the risk assessment is a prerequisite to an affirmative agency finding that an application is "safe enough." *Ethyl Corp.*, however, supports an assessment which is a cautious determination of danger in spite of inconclusive evidence, warranting action under a "precautionary" statute.[94] The fact that the NRC determination is a statutory prerequisite designed to ensure protection to the public should render its reasoning more accountable on review.[95]

risk limits is that implicit in Chairman Hendrie's comparative assessment. *Safety Study Hearings, supra* note 77.

90. A judicial lament has been raised by Judge McGowan of the United States Court of Appeals for the District of Columbia when faced with narrower technical issues. Union of Concerned Scientists v. AEC, 499 F.2d 1069, 1094 (D.C. Cir. 1975). This position has been disputed by the late Judge Leventhal of the same court who urged the judiciary to embrace the challenge presented to the legal system by issues of modern technology. Ethyl Corp. v. EPA, 541 F.2d 1, 68-69 (D.C. Cir. 1976), *cert. denied*, 426 U.S. 941 (1976) (concurring opinion).

91. Reserve Mining Co. v. EPA, 514 F.2d 492 (8th Cir. 1975); Ethyl Corp. v. EPA, 541 F.2d 1 (D.C. Cir. 1976), *cert. denied*, 426 U.S. 941 (1976).

92. 541 F.2d 1 (D.C. Cir. 1976), *cert. denied*, 426 U.S. 941 (1976).

93. 501 F.2d 722, 733, *cited in* Ethyl Corp. v. EPA, 541 F.2d 1, 23 (D.C. Cir. 1976), *cert. denied*, 426 U.S. 941 (1976).

94. Ethyl Corp. v. EPA, 541 F.2d 1, 28 (D.C. Cir. 1976), *cert. denied*, 426 U.S. 941 (1976).

95. The precautionary nature of the finding was a central factor in *Ethyl Corp.*

VII. CONCLUSION

Courts will accord less deference to the NRC's safety conclusions as finer techniques for risk assessment are developed.[96] Even without a demonstrably reliable methodology as a contrast to the NRC's practices, future events may command searching judicial attention.[97] It would be difficult to justify continued construction of nuclear reactors in the wake of further accidents of the magnitude of Three Mile Island.[98] In a very real sense the health and welfare of the industry, and of the affected communities, lie in the industry's own hands. The evidence and the findings that have been reported as a result of the Three Mile Island investigations could have a beneficial effect on the regulatory process.[99]

Serious problems are evident in the regulatory scheme. Open questions on issues involving the functioning and integration of fundamental components and systems should be answered, *before* the populace is subjected to uncertain risks and the solvency of vast utility networks is threatened. The classification of an issue as generic should not serve as a shield from effective agency and court review. The NRC has necessary corrections to make in the regulatory structure: at a minimum, opening USI's to adversarial examination on review, creating a clear standard of acceptable risk, developing an adequate methodology for assessing risk, and, generally, revaluating the current administrative ability to recognize and

"While awaiting . . . statistical certainty may constitute the typical mode of scientific behavior, its appropriateness is questionable in environmental medicine, where regulators seek to prevent harm that often cannot be labeled 'certain' until after it occurs." *Id.* at 25 n.52.

96. *See generally* Handler, *A Rebuttal: The Need for a Sufficient Scientific Base for Government Regulation*, 43 GEO. WASH. L. REV. 808 (1975).

97. For example, in Carolina Environmental Study Group v. United States, 510 F.2d 796 (D.C. Cir. 1975), the court found that the NRC had a reasonable basis to omit Class 9 accidents (the most serious variety) from its environmental considerations, relying on the NRC judgment that "the probability of an occurrence may be so low as to render it almost totally unworthy of consideration." *Id.* at 799. The NRC staff has concluded that TMI was a Class 9 accident. STAFF REPORT ON THE NRC, *supra* note 1, at 68-69 (quoting Pub. Serv. Elec. & Gas Co., Docket 50-272, at 3 (Salem Nuclear Generating Station, Unit 1; "NRC Staff Response to Board Question No. 4 Regarding the Occurrence of a Class 9 Accident at Three Mile Island")).

98. The cost of replacement at TMI, not including the loss of revenue to the surrounding community (estimated to be between $5.7 and 8.2 million) is estimated to range from $1 to 3 billion, depending on the type of replacement unit. 1 TECHNICAL TASK REPORT, *supra* note 56, at 76-82.

99. This note touched upon only a small part of the comprehensive review undertaken in those reports.

remedy licensing dangers. If the maintenance of the status quo appears to be the NRC's main concern, Congress should justify the judiciary's reliance on legislative oversight responsibility by mandating these fundamental changes. If these changes are not forthcoming, the judicial deference heretofore afforded NRC decision-making must be reevaluated.

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