UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
COMMONWEALTH EDISON COMPANY	)	Docket Nos. 50-454 OL
	)	50-455 OL
(Byron Nuclear Power Station,	)	
Units 1 & 2)	)	

COMMONWEALTH EDISON COMPANY'S PROPOSED FINDINGS OF FACT AND CONCLUSIONS OF LAW REGARDING CLASS 9 ACCIDENTS

July 1, 1983

DS03

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* See Applicant's May 31, 1983 submittal.

** See Applicant's June 7, 1983 submittal.

*** See Applicant's June 20, 1983 submittal.

**** See Applicant's June 24, 1983 submittal.

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* See Applicant's May 31, 1983 submittal.

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OF LAW REGARDING CLASS 9 ACCIDENTS*

OPINION

I. PROCEDURAL HISTORY

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II. CONTENTIONS

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G. Rockford League of Women Voters' ("League") Contentions 8 and 62 and DAARE/SAFE Contention 2a - Class 9 Accidents

The League contends in Contentions 8 and 62 that no meaningful assessment of the environmental risks associated with severe accidents at Byron Station has been performed and

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* These proposed findings are presented in the form of a partial initial decision which addresses one of the eight litigated issues, specifically, Class 9 accidents. The proposed findings on the other seven issues have been or will be submitted in accordance with the schedule stated in the "Procedural History" section of "Commonwealth Edison Company's Proposed Findings of Fact and Conclusions of Law Regarding Seismology, Waterhammer, and ALARA" filed on May 31, 1982, into which this document is incorporated.

that the design of Byron Station does not provide protection against such accidents. DAARE/SAFE asserts in Contention 2a that with the addition of Byron Station, the potential for cumulative dose effects from severe accidents at nuclear power plants in Northern Illinois provides an unreasonable level of risk to the health and safety of the public in that area.

Applicable Law

Section 102(2)(C) of the National Environmental Policy Act ("NEPA"), 42 U.S.C. §§ 4321-4361 (1976), requires the NRC to prepare a detailed environmental impact statement for any action that could significantly affect the quality of the environment. 42 U.S.C. § 4332(2)(C). Section 102(2) also requires the NRC to explore the environmental ramifications of its proposed actions "to the fullest extent possible." Compliance with Section 102(2)(C) and 10 C.F.R. Part 51 of the Commission's regulations is required before an operating license can be issued.

To satisfy the NRC's NEPA obligations under Section 102(2)(C), the NRC Staff, pursuant to 10 C.F.R. § 51.26, has the responsibility for preparing a final environmental statement ("FES".) The NRC Staff's FES must contain a discussion of certain types of accident scenarios as part of the required environmental review conducted for reactor licensing. Consumers Power Company (Midland Plant, Units 1 and 2), ALAB-123, 6 AEC 331, 346 (1973).

In its Statement of Interim Policy, dated June 13, 1980, on "Nuclear Power Plant Accident Considerations Under the National Environmental Policy Act of 1969," the Commission stated that each NRC Staff FES must include a discussion of the more severe kinds of very low probability accidents that are physically possible at the particular nuclear power plant which is the subject of the FES. 45 Fed. Reg. 40101, 40103 (June 13, 1980.) Such accidents were commonly referred to as Class 9 accidents following an accident classification scheme proposed in 1971 for purposes of implementing NEPA. These accidents are now generally referred to as "severe accidents."

The adequacy of the severe accident analysis, conducted in accordance with the Commission's Statement of Interim Policy, contained in the NRC Staff's FES for the Byron Station, is determined by a rule of reason. Columbia Basin Land Protection Association v. Schlesinger, 643 F.2d 585, 592-593 (9th Cir. 1981). This rule requires that, in order to satisfy minimal standards of adequacy under Section 102(2)(C), the NRC Staff's FES need only contain a reasonably thorough discussion of the significant aspects of the probable environmental consequences of the action being considered, not a discussion of every uncertainty pertaining to the probability and consequence of severe accidents. See Jicarilla Apache Tribe of Indians v. Morton, 471 F.2d 1275, 1280 (9th Cir. 1973); Environmental Defense Fund, Inc. v. Corps of Engineers, 348 F. Supp. 916, 927 (N.D. Miss. 1972); Long Island Lighting Co. (Shoreham Nuclear Power Station), ALAB-156, 6 AEC 831, 837-838 (1973).

Accordingly, if the severe accident analysis contained in the Byron FES has been prepared and issued in accordance with NEPA as elicited by the Commission's Interim Statement of Policy, supra, and if the NRC Staff has made a good faith effort in its FES to describe the reasonably foreseeable environmental impacts of the occurrence of such a severe accident at the Byron Station, the requirements of Section 102(2)(C) will be fully satisfied. Accord Scientists' Institute for Public Information, Inc. v. Atomic Energy Commission, 481 F.2d 1079, 1092 (D.C. Cir. 1973).

The Commission may not issue an operating license to a nuclear power plant until reasonable assurance exists that an applicant will conduct its activities under the license without endangering the health and safety of the public. 10 C.F.R. § 50.57(a)(3). Before such a finding can be made, applicants must demonstrate compliance with, inter alia, the Commission's regulations in 10 C.F.R. Parts 50 and 100 governing the evaluation of design basis accidents.

An applicant for an operating license must present, in the final safety analysis report ("FSAR"), a final analysis of the proposed reactor facility demonstrating the adequacy of the structures, systems and components provided for the prevention and/or mitigation of accidents. 10 C.F.R. § 50.34(b)(4). The FSAR information, generally chapter 15, is reviewed by the NRC Staff in accordance with a corresponding section in the Standard Review Plan. (Section 15 of NUREG-0800, Rev. 2, July 1981). Generally, the adequacy of the accident analyses in the

FSAR is measured against the design requirements of Appendix A to 10 C.F.R. Part 50, and with respect to consequences, against the guideline values set forth in 10 C.F.R. § 100.11. Id. Once compliance has been shown with these and other applicable regulations and absent any showing that these regulations are inadequate to protect the health and safety of the public, an applicant is entitled to an operating license as a matter of law. Maine Yankee Atomic Power Company (Maine Yankee Atomic Power Station), ALAB-161, 6 AEC 1003, 1010 (1973).

An applicant for an operating license also has the ultimate burden of proof in a licensing proceeding. 10 C.F.R. § 2.732. However, Commission adjudicatory decisions have stressed that in order to trigger the Applicant's duty to assume the ultimate burden of proof with respect to contentions, intervenors must first come forward on specific issues with sufficient evidence to require reasonable minds to inquire further. Consumers Power Company (Midland Plant, Units 1 and 2), ALAB-315, 3 NRC 101, 110-112 (1976); Metropolitan Edison Co. (Three Mile Island Nuclear Station, Unit No. 1), ALAB-697 16 NRC ___, 2 Nuc. Reg. Rep. (CCH) ¶30,738, ¶30,738.07 (October 22, 1982). Pennsylvania Power and Light Company and Allegheny Electric Cooperative, Inc. (Susquehanna Steam Electric Station, Units 1 and 2), ALAB-613, 12 NRC 317, 340 (1980); Consumers Power Company (Midland Plant, Units 1 and 2), CLI-74-5, 7 AEC 19, 30-32 (1974), reversed sub nom. Aeschliman v. NRC, 547 F.2d 622, 628 (D.C. Cir. 1976), reversed and remanded sub nom. Vermont Yankee Nuclear Power Corp. v. Natural Resources

Defense Council, Inc., 435 U.S. 519, 553-554 (1978). The duty of an intervenor to go forward with evidence to buttress its position is particularly applicable to novel and evolving issues or contentions raised before the Commission. Consumers Power Company (Midland Plant, Units 1 and 2), CLI-74-5, supra, 7 AEC at 31-32.

It is against the statutes, regulations and Commission decisions discussed above that the Licensing Board must weigh the evidence on the League and DAARE/SAFE Contentions 8, 62, and 2a.

Contention 8: Assessment
of Severe Accident Risks

The League contends that neither Applicant nor the NRC Staff has made a meaningful assessment of the environmental risks associated with Class 9 accidents at Byron Station. (Finding 362.) The term "Class 9," or "beyond design basis," accidents commonly describe accidents involving successive failures which are more severe than those postulated for the design basis for protective systems and engineered safety features installed in nuclear power reactors. This type of accident is historically postulated to involve significant physical deterioration of the fuel in the reactor core and of the containment structure. (Finding 373.) The consequences of such an accident could be severe, but the probability of the occurrence of such an accident is extremely small. (Finding 374.)

The term "Class 9 accident" originally was used in a proposed Annex to Appendix D to 10 C.F.R. Part 50. In its

June 13, 1980 Statement of Interim Policy, however, the Commission withdrew that proposed Annex and with it, the term "Class 9 accident." As a consequence, the term "severe accident" will be used hereafter in this Opinion.

It is undisputed that no evaluation of the environmental risks associated with severe accidents at Byron Station has been submitted by the Applicant on this record. However, the League is mistaken in the notion that such an evaluation is required of the Applicant. In its Interim Policy Statement with respect to Nuclear Power Plant Accident Considerations Under the National Environmental Policy Act of 1969, the Commission states clearly that environmental reports submitted by applicants for operating licenses must include severe accident evaluations _only_ if such documents are submitted after July 1, 1980. (Statement of Interim Policy, 45 Fed. Reg. at 40103 (June 13, 1980).) Our review of the docket in this case discloses that the Applicant submitted its environmental report to the NRC Staff on November 30, 1978. (See letter from Mr. Cordell Read to NRC's Mr. H.R. Denton.) Hence, the Applicant's environmental report need not be amended to include an evaluation of severe accidents.

In any event, the responsibility for the performance of a NEPA evaluation rests with the cognizant federal agency rather than with a private party, such as a license applicant. Accordingly, the NRC Staff has prepared an environmental impact statement for Byron, and it has evaluated, _inter alia_, the risks associated with severe accidents. (Finding 365.) Thus, the Board will construe Contention 8 as alleging that the NRC Staff's assessment of environmental risks from severe accidents, as presented in the Byron Final Environmental Statement ("FES"), is inadequate.

The League's principal criticism of the FES is that the NRC Staff relied upon a flawed methodology to assess the environmental risks of severe accidents at Byron Station. The NRC Staff used the probabilistic risk assessment ("PRA") methodology of the Reactor Safety Study ("WASH-1400"). (Finding 375.) The League contends that several studies have cast doubt on both the methodology and the conclusions of WASH-1400 thereby destroying the reliability of the document for any use, including as a source and reference tool for the NEPA evaluation of severe accidents. (Finding 362.)

One of the studies the League claims casts doubt on the use of the WASH-1400 methodology is the Lewis Committee Report. Dr. H. W. Lewis was chairman of a review group established by the Commission on July 1, 1977 for the purpose of providing an independent assessment of WASH-1400. The results of that assessment were published in September 1978 in a document entitled "Risk Assessment Review Group Report to the U.S. Nuclear Regulatory Commission," NUREG/CR-0400. The League interprets these results as discrediting WASH-1400.

Although the League provided no witnesses to support its position (Finding 372), witnesses for the Applicant and the NRC Staff concede that the Lewis Committee did criticize some aspects of WASH-1400. (Finding 418.) However, they stress that the criticisms did not challenge the correctness of the study's basic methodology. Moreover, these witnesses point out that the probabilistic methodology employed by WASH-1400 was endorsed by the Lewis Committee and that the Committee supported its use in quantifying the accident probabilities associated with nuclear power stations. (Finding 417.) The Board's own review of NUREG/CR-0400 concurs with that of the witnesses for the Applicant and the NRC Staff, facts that we take official notice of pursuant to 10 C.F.R. § 2.743(i). Thus, the Board concludes that the Lewis Committee Report does not discredit the basic PRA methodology used in WASH-1400.

Similarly, the League is mistaken in its assertion that the Commission withdrew its endorsement of WASH-1400. Although the Commission did withdraw its support of the Executive Summary of that study, it still remains committed to the use of the WASH-1400 PRA methodology as a regulatory tool. (Findings 419, 421, 422.) This position is supported by other independent committees and investigatory commissions. (Finding 420.)

The League also claims that the results of a recent study, the Precursors to Potential Severe Core Damage Accidents: 1969-1979 A Status Report (NUREG/CR-2497) (the "Precursor Study"), invalidate the use of WASH-1400's core melt frequency estimates because the Precursor Study estimated the frequency of severe core damage accidents to be significantly higher than WASH-1400's estimated core melt frequency. (Finding 423.) We disagree.

The authors of the Precursor Study estimated the frequency of severe core damage accidents, not the frequency of severe core melt accidents as was done in WASH-1400. Core damage accident involve damage to the geometric configuration of the core with little or no melting. Generally, such accidents present no challenge to the integrity of reactor containments and release very little radiation to the environment. Since the Commission's charter for WASH-1400 was to estimate the risk to the public from severe accidents in nuclear power plants, the authors of WASH-1400 deliberately postulated events that placed the public at risk, i.e., systems failures that led to core melt, containment failure and large releases of radioactivity to the environment. (Finding 427.) Thus, a comparison of the WASH-1400 core melt frequencies with core damage frequencies of the Precursor Study is not apt.

In addition, the results of the Precursor Study are unreliable because a flawed methodology was used. In that study, a rare event at a specific plant, such as the fire at Browns Ferry, was used as input to generic event trees. (Finding 424.) That methodology is incorrect because it does not take into account the specific plant event trees or the system failure probabilities applicable to that plant. (Finding 425.) In fact, the Institute for Nuclear Power Operations has applied those same specific events to event trees for the plant at which the event occurred and obtained estimates of core damage frequencies much lower than those reported in the Precursor Study. (Finding 426.)

The Board finds that the Commission, as well as studies by peer groups like the Lewis Committee Report, support the use of the PRA methodology of WASH-1400. (Findings 417, 419, 420, 421, 422.) Furthermore, the Board finds for the reasons indicated above that reliable judgments cannot be drawn from the results of the Precursor Study. (Findings 424, 425, 426, 427.) The League has therefore not cited any credible evidence which questions the validity of WASH-1400's methods. On the basis of the above evidence, the Board finds that the WASH-1400 PRA methodology is an appropriate tool for use in assessing the environmental risks of severe accidents at Byron Station.

We turn now to an examination of the NRC Staff's evaluation of severe accidents in the FES. As a first step, the NRC Staff adopted from WASH-1400 the four sets of accident sequences determined to represent the risk at a pressurized water reactor ("PWR") like Byron Station. (Findings 376, 377, 385.) Although those accident sequences recently were rebaselined to incorporate better data and analytical techniques developed after WASH-1400 was published, the rebaselined results have only minor differences from WASH-1400's predictions. (Findings 377, 378, 379.)

The NRC Staff used these four rebaselined accident sequences as its starting point in preparing its probabilistic assessment of severe accidents in the FES. (Finding 380.) Of the four events, the largest risk contributor is the Event V accident sequence. In general, this sequence assumes that events occur which cause the low pressure injection subsystem ("LPIS") to rupture which, in turn, causes a loss of coolant accident ("LOCA"). It is then assumed that the LPIS rupture and the LOCA environment cause the LPIS makeup function to fail entirely, and core damage occurs. (Finding 381.)

The second accident sequence, TMLB', assumes that the station loses all of its ac power sources and an independent failure of its diesel driven auxiliary feedwater train occurs. This would result in plant trip, and the station would be unable to remove heat from the reactor core. The heat would ultimately cause the core to uncover and melt and could cause the failure of the containment. (Finding 382.)

The last two accident sequence designators are called PWR 3 and PWR 7. PWR 3 assumes a small LOCA occurs in the reactor vessel cavity and the containment spray system in the recirculation mode fails. This will result in an overpressure failure of the containment. The sequence also assumes that the emergency core cooling system fails and that the core melts in a breached containment. (Finding 383.) PWR 7 consists of several accident sequences that involve a containment base mat melt-through. All containment engineered safety features are assumed to operate until the base mat is penetrated. (Finding 384.)

The NRC Staff first estimated the probabilities of each of these four accident sequences occuring because risks are determined by displaying both the probability of an event and its consequences. (Finding 386.) For Byron, the NRC Staff estimated that there are two chances in one million per reactor-year that an Event V, the largest risk contributor, will occur. PWR 7 was estimated to have four chances out of every one hundred thousand reactor-years of occurring and is the most probable of any of the four events. (Finding 387.)

The NRC Staff also determined the total dose and health impacts from atmospheric releases of radioactive materials from severe accidents, and presented both the consequences of each accident and its probability. (Finding 388.) These calculations show that for atmospheric releases, there is one chance in one million per reactor-year that thirty-seven early fatalities and 3990 latent cancer fatalities will occur due to the operation of Byron Station. (Finding 390.)

The NRC Staff also evaluated the potential environmental impacts of a release of radiation to the groundwater. The NRC Staff did this by comparing the results from the generic small river land-based site in the Liquid Pathway Generic Study ("LPGS") with the liquid pathway consequences of postulated severe accidents at Byron Station. (Findings 391, 393.) The potential consequences of a release of radioactivity to liquid pathways at Byron were analyzed in the FES, and it was determined that potential doses from such a release are of the same order of magnitude as that predicted for the corresponding small river site evaluated in the LPGS. (Finding 394.) Thus, it may be concluded that, for Byron, the contribution to dose and health impacts from radioactive releases to the groundwater is very small in comparison with corresponding contributions from atmospheric releases.

The matter of the adequacy of the NRC Staff's liquid pathway analysis is treated extensively elsewhere in this Opinion in connection with the Board's consideration of League Consolidated Contentions 39 and 109. Having found in favor of the NRC Staff's environmental analysis with respect to those contentions, the Board concludes, a fortiori, that the potential releases of radioactivity to the groundwater contribute very little to the total dose and health impacts from severe accidents at Byron Station. (Finding 395.)

The NRC Staff also calculates the average values of risk associated with severe accidents at Byron Station by summing the probabilities multiplied by the consequences over the entire distribution range. (Finding 397.) These include an average value of 0.00026 for early fatalities and 0.0125 for latent cancer fatalities exclusive of the thyroid. The average value for latent thyroid cancer fatalities is 0.0035. (Finding 398.) The NRC Staff concludes in the FES that, assuming protective action is taken, the environmental risks of severe accidents is roughly comparable to the risks associated with the normal operation of Byron Station and other human activities. (Findings 399, 401, 402.)

Mr. Saul Levine, Applicant's witness, performed an independent assessment of the NRC Staff's severe accident evaluation. (Finding 403.) Mr. Levine is a highly credible witness who has been actively involved in the field of nuclear energy for almost 30 years and was Project Staff Director for WASH-1400. Mr. Levine emphasized that the methods used in the

Byron FES are consistent with the approach used in environmental impact statements for other nuclear power plants and that results of PRAs performed for other plants suggest that the values reported in the Byron FES are conservative. (Finding 404.) He concluded that the methods used in the FES to evaluate the risks of severe accidents are reasonable and that the estimated risks of those accidents are conservative. (Finding 403.) Mr. Levine only disagreed with the NRC Staff on the quantification of uncertainties in the FES. (Findings 411, 413.)

The NRC Staff had initially limited its discussion of the uncertainties involved in the risk assessment of severe accidents in the FES to a qualitative evaluation. (Finding 407.) However, in their supplemental testimony, the NRC Staff witnesses stated that the accident risks from the summation of internal and external causes, exclusive of sabotage, could be as much as factors of 10 and 30 higher than the risks from internal events alone, as presented in the FES. The NRC Staff witnesses based this determination on the risk multipliers computed for Indian Point and Zion through site-specific PRAs. (Findings 408, 409.) The NRC Staff witnesses also testified at the hearings that the uncertainties associated with the risks of accidents from both internal and external events, excluding sabotage, are less than 100 times the estimation in the FES of risks from internal events alone. (Finding 410.)

Mr. Levine disagreed with the NRC Staff's approach for quantifying uncertainties. First, he pointed out that the relationship of internal to external event contributors to risk is dependent on the specific plant design and the specific site involved. In Mr. Levine's view, this axiom invalidates the NRC Staff's application of the Indian Point and Zion risk multipliers to Byron absent any convincing evidence showing otherwise. (Finding 411.) We find this judgment difficult to dispute. The threat to plant safety from any given external event will obviously vary from site to site and from plant to plant. Thus, the Board finds that insufficient evidence exists to quantify reasonably the ratio of internal to external events at Byron. Nor for the reasons explained below do we believe it necessary to do so.

The NRC Staff used a very general approach, rather than a plant-specific PRA, to prepare the FES. Greater precision is unnecessary to formulate a judgment as to whether the environmental risks associated with postulated accidents at nuclear power plants are small, moderate or large. (Finding 414.) Although a general approach is adequate to assess the environmental risks to the public from potential severe accidents, its application is inappropriate to determine uncertainty estimates. Such estimates can only be determined meaningfully by a plant-specific analysis. (Finding 412.)

In addition, a quantification of accident risk estimates, as Mr. Levine explained, is not needed because the estimated risks in the FES are sufficiently conservative so that uncertainties will not affect the overall conclusion that the predicted public risks from severe accidents are small when compared to other risks. (Finding 413.) Simply stated, these uncertainties are offset by the many conservative assumptions the NRC Staff made in preparing the FES.

It was suggested during the cross-examination of Mr. Levine (Tr. 6959-6961) that the quantification of risk uncertainties is required by the Commission's June 13, 1980 Statement of Interim Policy. Although the Policy Statement does require that such uncertainties be recognized in the FES, a quantitative assessment is not mandated. (Statement of Interim Policy, 45 Fed. Reg. at 40103 (June 13, 1980).) Indeed, previous environmental impact statements contained only a qualitative characterization of these uncertainties. (Levine, Tr. 6992.)

The Board commends the NRC Staff for its conscientious attempt to provide more certainty in its FES evaluation. However, for the reasons explained by Mr. Levine, meaningful quantification of the uncertainties associated with the NRC Staff's severe accident assessment is neither possible given the non plant-specific approach used in the FES nor necessary because of the many conservatisms in the evaluation. Therefore, the Board finds that consistent with the Commission's past practice, the June 1980 Policy Statement and the "rule of reason" articulated by the Courts, that the quantification of risk estimate uncertainties is unnecessary, and that the assessment set forth in the FES is adequate.

The League developed only one argument that attacked the adequacy, rather than the methodology, of the assessment in the FES of the risks from severe accidents. The League contends that additional considerations of the risks of core-melt accidents are needed at Byron Station because the site is located in a highly populated area. (Finding 362.) That assertion is, however, not true as the population density surrounding the Byron site is generally much less than the average population density of existing and proposed nuclear power reactor sites. (Finding 428.) Thus, the population surrounding Byron does not justify additional considerations of the risks from severe accidents.

The League, in Contention 8, refers to a number of documents in support of its position, i.e., NRC studies that are either unpublished or not common public knowledge, a GAO report, and a letter apparently written by Mr. Case. (Finding 362.) The League, however, neither offerred the documents into evidence nor presented any testimony concerning them. Indeed, no mention of these documents is made on this record except for Contention 8 itself. In this circumstance, the Board finds that the League has failed to satisfy its initial burden of establishing, prima facie, that those documents could conceivably show that the severe accident evaluation in the FES is inadequate. The Board therefore will not consider this matter further.

Based on the evidence, the Board finds that the Byron FES presents an adequate assessment of the environmental risks to the public from severe accidents at Byron Station. The NRC Staff used acceptable methods in preparing the FES to quantify the severe accident probabilities and consequences associated with the operation of Byron Station. The use of that methodology for this purpose has not been called into question. Furthermore, the FES describes in detail the procedures used to calculate the probabilities, the dose and health impacts, and the risks that may occur as a result of a severe accident, as well as the uncertainties involved in the assessment. The Board finds, contrary to the claim in Contention 8, that the NRC Staff has performed an adequate and meaningful assessment of the risks from severe accidents at Byron Station.

Contention 62: Accident Mitigation

The League contends that the design of Byron Station does not provide adequate protection against Class 9 accidents. The NRC process for making safety evaluations and regulatory decisions with respect to nuclear power reactors is called "deterministic" because the elements that must be considered are determined by qualitative engineering judgment instead of by quantitative probabilistic estimates. (Finding 432.) PRAs are neither used by the NRC Staff for nuclear power plant safety evaluations nor are they required for such evaluations. (Findings 434, 435.)

The licensing process provides protection in that it requires that several specifically defined accidents, called design basis accidents, be analyzed to demonstrate that the plant does not subject the public to an undue risk from radiation. This process results in the installation of accident prevention and mitigation systems. (Finding 431.) For example, Byron Station is required to have a reactor protection system, emergency core cooling system, special containment building design features, containment sprays and fan coolers, and an auxiliary feedwater system. All of these features protect against the occurrence, or mitigate the consequence, of a severe accident. (Finding 440.)

Furthermore, Byron Station has other design and operating features that provide protection against the occurrence of a severe accident. Many of these features were instituted as a result of the lessons learned from the accident at Three Mile Island. (Finding 436.) These features include a reactor vessel head vent system, additional monitoring instruments, and improved emergency response facilities. (Finding 437.) In addition, all licensees must now review the adequacy of many of its procedures and the capabilities of its employees. (Finding 438.)

In sum, the deterministic approach to evaluating safety, which relies on the experience and wisdom of skilled engineering professionals, has resulted in good safety records for nuclear power plants, including Byron Station. (Findings 432, 433.) Based on the uncontroverted evidence, the Board finds that Byron Station's design provides, consistent with the requirements of the Commission's regulations, adequate protection against severe accidents.

Contention 2a: Incremental
Risk From Byron Station

DAARE/SAFE contends that, with the addition of Byron Station, the potential for cumulative dose effects from accidents at nuclear power plants in Northern Illinois poses an unreasonable level of risk to the health and safety of the Byron Station area residents. (Finding 364.) This contention has two basic components. In essence, it alleges that the probability of a severe accident occurring at more than one plant in the same general area, coupled with the health risks to the public from exposure to radiation from more than one severe accident, impose an unreasonable level of risk to the public's health and safety.

As to the first aspect of this contention, the probability of a core melt accident at any one nuclear power plant is generally about one in ten thousand per reactor-year. (Finding 442.) The probability of more than one severe accident occurring at more than one nuclear station, with significant cumulative radiological effects to people in the same area, is significantly smaller. (Finding 443.)

With respect to the second component of this contention, PRA evaluations have shown that the accident risk of an early fatality to people living fifteen miles or more from a nuclear station is extremely small. Thus, no increased risk of early fatalities exists from the coupling of the risks from the plants located in Northern Illinois. (Finding 446.) Studies also show that the probability of a latent cancer fatality caused by exposure to radiation from a severe nuclear station accident is negligibly small when compared to the probability of an individual dying from cancer contracted from other sources. Thus, the number of nuclear reactors located in an area is inconsequential to an individual's latent cancer fatality risk. (Finding 447.)

Based on the uncontroverted evidence, the Board finds that with the addition of Byron Station, the cumulative dose effects from accidents at nuclear power plants in the northern Illinois area do not pose an unreasonable level of risk to the health and safety of the public. The probability that two or more severe accidents will occur in the same area is extremely small, and the number of nuclear plants in a given area has an inconsequential effect on an individual's early fatality and latent cancer fatality risks. Thus, the possibility of cumulative doses from accidents at more than one nuclear power plant in an area does not endanger the public health and safety.

Conclusion

The Board finds that an adequate assessment of the environmental risks from severe accidents at Byron is presented in the FES. This assessment was made by using the acceptable methodology and clearly considers the probabilities of the occurrence of a severe accident and the probabilities of the consequences of such an accident. Furthermore, the FES presents a qualitative assessment of the uncertainties involved in its risk estimation. Accordingly, the Board concludes that the requirements of 10 C.F.R. § 51.20(a), NEPA, and the Commission's June 13, 1980 Statement of Interim Policy have been satisfied.

The Board also finds that the requirements of 10 C.F.R. § 50.57(a)(3) and 10 C.F.R. Part 100 have been met. Byron has numerous design features and operating procedures, made before and after the TMI accident, which protect against the occurrence of a severe accident. Reasonable assurance exists that Byron can be operated without endangering the health and safety of the public.

Finally, the Board finds that the possibility of cumulative doses to residents of Northern Illinois from severe accidents at more than one nuclear power station does not create an undue risk to the public health and safety. The possibility of a severe accident at more than one station in the same area is exceedingly small, and the number of reactors in an area is inconsequential to an individual's early fatality or latent cancer fatality risk.

Accordingly, it is ordered that Contentions 8, 62, and 2a are dismissed.

FINDINGS OF FACT

III. Contentions

* * * * * *

G. League Contentions 8 and 62 and DAARE/SAFE Contention 2a - Class 9 Accidents

362. League Contention 8 reads as follows:

> 8. Neither C.E. nor the Staff has presented a meaningful assessment of the risks associated with the operation of the proposed Byron nuclear facility, contrary to the requirements of 10 C.F.R. § 51.20(a) and § 51.20(d). Studies carried out by the NRC have identified accident mechanisms, considered credible, which would lead to uncontrollable accidents and release to the environment of appreciable fractions of a reactor's inventory of radioactive materials. Traditionally, these accident potentials have been downplayed or ignored on the basis of the Rasmussen Report. However, the Lewis Committee has now called into serious question the entire methodology, as well as the findings and conclusions, of the Rasmussen Report, which led the NRC to withdraw official reliance on the Rasmussen Report, yet the Staff still regulates upon the validity of the basic conclusions therein. In addition, NRC Staff studies, which are not common public knowledge, have cast doubt upon numerous of the specific conclusions of the Rasmussen Report. For example, in one secret NRC study, estimates of the "killing distance" were made, referring to the range over which lethal injuries would be received under varying weather conditions from the release of radioactive material in a nuclear power plant accident. Depending upon prevailing weather conditions, this "killing distance" was estimated to be up to several dozen miles from the accident-damaged reactor. Unpublished document from Brookhaven National Laboratory, USAEC. In addition, the Liquid Pathways Study, NUREG-0440 (February, 1978), highlights the incomplete safety assessment currently performed by the NRC, particularly with respect to incomplete review of all credible accident sequences. A General Accounting Office report pertaining to that study criticizes the NRC's failure to

consider core-melt accidents in assessments of relative differences in Class 9 risks. The March 7, 1978 letter from the NRC's Mr. Case to the Commissioners (Secy-78-137) also urges the inclusion of core-melt considerations in site comparisons in the case of sites involving high population density, such as Byron and the surrounding area in which live now (or at time of proposed operation) upwards of 500,000 persons. Moreover, neither C.E. nor the NRC Staff has presented an accurate assessment of the risks posed by operation of Byron, contrary to the requirements of 10 C.F.R. § 51.20(a) and § 51.20(d). The decision to issue the Byron construction permit did not, and the presently filed analysis of C.E. and the Staff do not, consider the consequences of so-called Class 9 accidents, particularly core meltdown with breach of containment. These accidents were deemed to have a low probability of occurrence. The Reactor Safety Study, WASH-1400, was an attempt to demonstrate that the actual risk from Class 9 accidents is very low. However, the Commission has stated that it "does not regard as reliable the Reactor Safety Study's numerical estimate of the overall risk of reactor accident." (NRC Statement of Risk Assessment and the Reactor Safety Study Report (WASH-1400) in Light of the Risk Assessment Review Group Report, January 18, 1979). The withdrawal of NRC's endorsement of the Reactor Safety Study and its findings leaves no technical basis for concluding that the actual risk is low enough to justify operation of Byron.

363. League Contention 62 reads as follows:

62. The design of Byron does not provide protection against so-called "Class 9" accidents. There is no basis for concluding that such accidents are not credible. Indeed, the Staff has conceded that the accident at TMI falls within that classification. Therefore, there is not reasonable assurance that Byron can be operated without endangering the health and safety of the public. See also Contention 8, supra.

364. DAARE/SAFE Contention 2a reads as follows:

2a. Due to the concentration of nuclear power plants already in Northern Illinois; the Applicant's record of incidents and violations in existing plants which have emerged since the granting of a Construction License for Byron;

> and the credibility which must now be given to large scale accident scenarios since TMI, Intervenors contend that the addition of Byron Station operations places an undue and unfair burden of risk from exposure to radioactive materials from accidental release on DeKalb-Sycamore and Rockford area residents. With the addition of two more nuclear power units in operation at Byron, the potential for cumulative dose effects from discrete accident events at plants in Northern Illinois under unfavorable meteorological conditions poses an unreasonable level of risk to the health and safety of DeKalb-Sycamore and Rockford area residents.

365. The primary evidence addressing Contention 8 is the NRC Staff's Final Environmental Statement ("FES") which assesses the environmental risks of severe accidents at Byron Station. (FES, NRC Staff Exhibit 2 at 5-44 through 5-67, §§5.9.4.5(2)-(7), 5.4.9.6.) The FES's discussion on severe accident risks was supplemented by testimony from NRC Staff witnesses L.G. Hulman, Branch Chief of the Accident Evaluation Branch; Millard L. Wohl, a nuclear engineer in the Accident Evaluation Branch; and Scott Newberry, a risk analyst in the Reliability and Risk Assessment Branch. (NRC Staff Prepared Testimony at 1-13, ff. Tr. 2091.) These witnesses adopted the evaluation in the FES of severe accidents as their testimony (Wohl, Hulman, NRC Staff Prepared Testimony at 2, ff. Tr. 2091) and testified as to its adequacy. (Wohl, Hulman, Newberry, NRC Staff Prepared Testimony at 2-13, ff. Tr. 2091.)

366. Applicant presented Saul Levine who conducted an independent evaluation of the NRC Staff's assessment of the risks in the FES and testified on his conclusions. (Levine, Applicant Prepared Testimony at 16-18, ff. Tr. 1930.) Mr.

Levine's testimony also supported the methodology used by the NRC Staff to prepare the FES. (Levine, Applicant Prepared Testimony at 3-16, ff. Tr. 1930.)

367. During the hearing, the NRC Staff revised its original testimony regarding the consideration of external events and the quantification of uncertainties associated with the risk estimates presented in the FES. (Hulman, Wohl, NRC Staff Prepared Testimony at 3-4, ff. Tr. 2091; Tr. 2089.)

368. In response to this quantification, Applicant filed rebuttal testimony, also prepared by Mr. Levine, that questioned the need for, and the meaningfulness of, quantitative uncertainty estimates on the conservative results presented in the FES. (Levine, Applicant Prepared Rebuttal Testimony passim, ff. Tr. 6956.)

369. In regard to Contention 62, Mr. Wohl of the NRC Staff testified on those design and operating procedure modifications, that have been implemented at Byron Station in response to the accident at Three Mile Island ("TMI"), that protect against severe accidents. (Wohl, NRC Staff Prepared Testimony at 14-20, ff. Tr. 2091; FES, NRC Staff Exhibit 2 at 5-40, §5.9.4.4(1) and 5-67, §5.9.4.6.)

370. Mr. Levine testified on post-TMI modifications and on the specific features of the design of Byron Station that provide protection against, or mitigate the consequence of, the occurrence of a severe accident. (Levine, Applicant Prepared Testimony at 18-22, ff. Tr. 1930.)

371. For Contention 2a, NRC Staff witness Dr. Edward F. Branagan, Jr., a health physicist in the Radiological Assessment Branch, testified on the potential radiological impacts of accidents at Byron Station, and Mr. Wohl testified on the possibility of the public receiving cumulative doses of radioactivity from accidents at more than one plant in the northern Illinois area. (Wohl, Branagan, NRC Staff Prepared Testimony at 20-22, ff. Tr. 2091.) Mr. Levine testified on both the probability and the consequences of severe accidents at two or more nuclear power plants in the same general area. (Levine, Applicant Prepared Testimony at 22-26, ff. Tr. 1930.)

372. Neither Intervenor presented any witnesses in support of these contentions.

373. "Class 9" or "beyond design-basis" accidents commonly describe accidents involving successive failures which are more severe than those postulated for the design basis for protective systems and engineered safety features. (Wohl, NRC Staff Prepared Testimony at 13, ff. Tr. 2091; Hulman, Tr. 2160.) This type of accident is historically postulated to involve significant physical deterioration of the fuel in the reactor core and deterioration of the containment structure's ability to limit the release of radioactive materials to the environment. (FES, NRC Staff Exhibit 2 at 5-44, §5.9.4.5(2).)

374. The consequences of such an accident could be severe, but the probability of the occurrence of such an accident is extremely small. (Wohl, NRC Staff Prepared Testimony at 13, ff. Tr. 2091; FES, NRC Staff Exhibit 2 at 5-44, §5.9.4.5(2).)

Contention 8: Assessment of Severe Accident Risks

375. In preparing Sections 5.9.4.5(2)-(7) and 5.4.9.6 of the FES, the NRC Staff evaluated the risks associated with severe accidents by using the probabilistic risk assessment ("PRA") methodology of the Reactor Safety Study ("WASH-1400"). (Wohl, NRC Staff Prepared Testimony at 6, ff. Tr. 2091; FES, NRC Staff Exhibit 2 at 5-44, §5.9.4.5(2).)

376. WASH-1400 found that four sets of accident sequences represent the risk associated with a pressurized water reactor ("PWR") like Byron Station. (FES, NRC Staff Exhibit 2, Appendix E at E-1 through E-2.)

377. The four sets of accident sequences adopted from WASH-1400 by the NRC Staff have been updated, or "rebaselined", to incorporate peer group comments and better data and analytical techniques developed after WASH-1400 was published. (FES, NRC Staff Exhibit 2 at 5-44, §5.9.4.5(2).)

378. The rebaselining evaluated individual accident sequences rather than evaluating many accident sequences grouped together into synthetic release categories as was done in WASH-1400. (FES, NRC Staff Exhibit 2, Appendix E at E-1 through E-2; Wohl, Tr. 2283.)

379. Only small overall differences exist between the results of the rebaselining and WASH-1400. (FES, NRC Staff Exhibit 2, Appendix E at E-1 through E-2.)

380. The NRC Staff used the four rebaselined accident sequences and release categories as its starting point in

preparing its probabilistic assessment of severe accidents in the FES. (FES, NRC Staff Exhibit 2 at 5-44, §5.9.4.5(2).)

381. The largest risk contributor from the rebaselined WASH-1400 PWR design is an accident sequence designated as "Event V". This sequence assumes that the multiple check valve barriers that separate the high pressure reactor coolant system from the low pressure injection system ("LPIS") fail in various modes and suddenly expose the LPIS to high overpressures and dynamic loadings. If this occurs, a high probability exists of LPIS rupture. This would cause a loss of coolant accident ("LOCA") which would bypass the containment and the mitigating features within the containment. It is then assumed that the LPIS rupture and the LOCA environment cause the LPIS makeup function to fail entirely. As a result, core damage would occur. (FES, NRC Staff Exhibit 2, Appendix E at E-2 through E-3.)

382. The second accident sequence, TMLB', assumes that the station loses all of its offsite and onsite ac power sources. This would result in plant trip and the inability of the station to use its normal method of removing heat from the reactor core. The removal of this heat would then require the operation of the diesel driven auxiliary feedwater train, but this accident sequence also assumes an independent failure of that system. The result is that the shutdown heat cannot be removed, and it will ultimately cause the core to uncover and melt. If ac power cannot be restored in time, the containment could fail due to overpressure which would result in large

releases of radioactive material from the containment. (FES, NRC Staff Exhibit 2, Appendix E at E-3.)

383. The third accident sequence, PWR 3, involves a series of failures. The series includes a small LOCA in the reactor vessel cavity with the failure of the containment spray system in its recirculation mode due to a lack of water in the containment sump. It is assumed that this failure coincident with a small LOCA will result in an overpressure failure of the containment. The sequence then assumes that the emergency core cooling system ("ECCS") fails due to mechanical loads or pump cavitation. The core is then assumed to melt in a breached containment which would lead to a significant release of radioactive materials. (FES, NRC Staff Exhibit 2, Appendix E at E-3 through E-4.)

384. The last accident sequence is referred to as PWR 7 and consists of several accident sequences, all of which involve a containment base mat melt-through. The sequences also assume that all containment engineered safety features will continue to operate as designed until the base mat is penetrated. This results in radioactive materials being released into the ground with some leakage through the ground to the atmosphere. (FES, NRC Staff Exhibit 2, Appendix E at E-4.)

385. The NRC Staff concludes in the FES that the four accident sequences described above represent the environmental risks associated with severe accidents at Byron Station. (FES, NRC Staff Exhibit 2 at 5-44, §5.9.4.5(2) and Appendix E at E-2.)

386. The risks associated with severe accidents at Byron Station are composed of two separate calculations: the probability, or frequency, that such an accident can occur and the consequences, or impact, of the accident if it did occur. The risks are determined by displaying both the probability of an event and its consequences. (FES, NRC Staff Exhibit 2 at 5-59, §5.9.4.5(6).)

387. Table 5.11 of the FES lists the fraction of core inventory released and the following frequencies per reactor-year for each of the four accident sequences: Event V, 2.0×10^{-6}; TMLB', 3.0×10^{-6}; PWR 3, 3.0×10^{-6}; and PWR 7, 4.0×10^{-5}. (FES, NRC Staff Exhibit 2 at 5-45, §5.9.4.5(2).)

388. Calculations of dose and health impacts or consequences from accidental atmospheric releases at Byron Station are charted in Figures 5.6 through 5.9 of the FES, in the form of probability distributions, and are summarized in Table 5.12. The four accident sequences and release sequences contribute to the results, and the consequences of each are weighted by the corresponding probability. (FES, NRC Staff Exhibit 2 at 5-48, §5.9.4.5(3).)

389. Table 5.12 shows that from atmospheric releases the probability per reactor-year is 5×10^{-6} that 7,000 people would be exposed to over 25 rems of radiation, that the population within 80 kilometers of the accident would be exposed to 1.5 million person-rems and the total population would be exposed to 12 million person-rems, that 180 latent cancer fatalities would appear in the population within 80 kilometers

of the site and 840 latent cancer fatalities would develop in the total population, and that the cost of offsite mitigating actions would be $430 million. No early fatalities or people being exposed to over 200 rems would occur. (FES, NRC Staff Exhibit 2 at 5-54, §5.9.4.5(3).)

390. Table 5.12 shows that from atmospheric releases there is one chance in a million per reactor-year that thirty-seven early fatalities and 3990 latent cancer fatalities will occur due to the operation of Byron Station. (FES, NRC Staff Exhibit 2 at 5-54, §5.9.4.5(3).)

391. The potential environmental impacts of accidental releases of radioactivity to the groundwater at Byron Station were evaluated in the FES by using the Liquid Pathway Generic Study ("LPGS"). (FES, NRC Staff Exhibit 2 at 5-56, §5.9.4.5(5).)

392. The NRC Staff determined in the LPGS that the potential release of radioactivity to the groundwater at generic sites and the resulting doses range from small fractions to very small fractions of those that might arise from the atmospheric pathway. (FES, NRC Staff Exhibit 2 at 5-56, §5.9.4.5(5).)

393. The NRC Staff determined that the results from the generic small river land-based site in the LPGS were an apt comparison to the liquid pathway consequences of a postulated accident at the Byron site. (FES, NRC Staff Exhibit 2 at 5-59, §5.9.4.5(5).)

394. The NRC Staff determined in the FES that the contribution to population dose from releases to the groundwater at the Byron site is on the same order of magnitude as that predicted for the corresponding small river site evaluated in the LPGS. (FES, NRC Staff Exhibit 2 at 5-59, §5.9.4.5(5).)

395. The contribution to dose and health impacts from potential radioactive releases to the groundwater from postulated severe accidents at Byron is very small in comparison with the corresponding contributions from atmospheric releases. (Levine, Tr. 1956-1957; FES, NRC Exhibit 2 at 5-56 through 5-59, §5.9.4.5(5).)

396. Table 5.13 of the FES lists the average values of risk associated with population dose, acute and latent fatalities, and cost for protective actions from all accidents at Byron Station. (FES, NRC Staff Exhibit 2 at 5-60, §5.9.4.5(6).)

397. The consequences of both atmospheric and groundwater releases and the risks associated with design-basis accidents were considered in preparing Table 5.13. The average values in the table were derived by summing the probabilities multiplied by the consequences over the entire range of distribution. (FES, NRC Staff Exhibit 2 at 5-59 through 5-60, §5.9.4.5(6).)

398. The average values of environmental risk due to accidents, per reactor year, are as follows: (i) 37 person-rems exposure to the population within 80 kilometers and 218 person-rems exposure to the total population; (ii) 0.0125 latent cancer fatalities for all organs except thyroid and

0.0035 latent cancer fatalities for only the thyroid; (iii) 0.00026 early fatalities; and, (iv) $8,400, in 1980 dollars, for protective actions and decontamination. (FES, NRC Staff Exhibit 2 at 5-60, §5.9.4.5(6).)

399. The population exposures and latent cancer fatality risks (items (i) and (ii) in Finding 398), assuming protective action is taken, are comparable to those for normal operation, excluding exposure to plant personnel. (FES, NRC Staff Exhibit 2 at 5-59 through 5-60, §5.9.4.5(6) and Appendix C.)

400. The experience from normal reactor operation cannot serve as a comparison with respect to early fatalities since none is associated with such operation. However, the risk from early fatalities may be compared to such risks from other human activities. (FES, NRC Staff Exhibit 2 at 5-60, §5.9.4.5(6).)

401. The risk of early fatalities from all potential accidents at Byron are small in comparison with the risk of early fatalities from other human activities. (FES, NRC Staff Exhibit 2 at 5-60, §5.9.4.5(6).)

402. The overall environmental risk of severe accidents at Byron Station is the sum of the comparisons described in Findings 399 and 401. (FES, NRC Staff Exhibit 2 at 5-66 through 5-67, §5.9.4.6.)

403. An independent assessment of the evaluation in the FES of the risks associated with severe accidents has been made. That evaluation determined that the approach in the FES

is reasonable and that the estimated risks in the FES are conservative. (Levine, Applicant Prepared Testimony at 16-18, ff. Tr. 1930.)

404. It was determined in the independent assessment that the calculations of environmental risks from severe accidents in the Byron FES are reasonable because PRAs performed for other plants support the methodology and the conclusions of the Byron FES and suggest that the values reported in it are conservative. (Levine, Applicant Prepared Testimony at 17, ff. Tr. 1930.)

405. It was also concluded in the independent assessment that the FES strikes a balance between conservatisms and uncertainties on the side of conservatism. Examples of conservatisms in the FES are: the core fission product inventory release fractions used in the FES probably are too large; the likelihood of steam explosions probably is much lower than what the FES assumes; the containment would require a longer time to fail than what the FES assumes; penetration of the basemat by the molten core would take longer than what is assumed in the FES (Levine, Applicant Prepared Testimony at 16-18, ff. Tr. 1930; Levine, Tr. 1970); and the probability assumed in the FES of an Event V occurring is too large. (Levine, Applicant Prepared Rebuttal Testimony at 6, ff. Tr. 6956.)

406. Uncertainties in the calculation of the accident sequence frequencies and of the consequences exist. (FES, NRC Staff Exhibit 2 at 5-45, §5.9.4.5(2) and 5-65 through 5-66, §5.9.4.5(7); Levine, Applicant Prepared Testimony at 17-18, ff. Tr. 1930.)

specific plant design and the site involved. (Levine, Applicant Prepared Rebuttal Testimony at 2-3, ff. Tr. 6956; Levine, Tr. 6970-6972.) No evidence was presented that showed that the ratios adopted by the NRC Staff from the Zion and Indian Point PRAs applied to Byron.

412. The very general approach used by the NRC Staff in the FES to calculate severe accident risks does not lend itself to meaningful uncertainty estimates because the evaluation in the FES is not plant-specific. Such estimates can only be determined meaningfully by a plant-specific analysis. (Levine, Applicant Prepared Rebuttal Testimony at 3-4, ff. Tr. 6956; Levine, Tr. 6990-6993.)

413. Mr. Levine testified that quantification of the uncertainties is unnecessary because the estimated risks in the FES are sufficiently conservative so that uncertainties will not affect the overall conclusion that the predicted public risk from severe accidents is small when compared to other risks. (Levine, Applicant Prepared Rebuttal Testimony at 4, ff. Tr. 6956.)

414. Greater precision is unnecessary to assist in formulating a judgment as to whether the environmental risks associated with accidents at nuclear power plants are small, moderate, or large. (Levine, Applicant Prepared Testimony at 18, ff. Tr. 1930.)

415. Based on the evidence, the Board finds that it is not possible to meaningfully assess the uncertainties in the Byron FES risk estimates and it is unnecessary to include

quantified uncertainty estimates in the already conservative calculations presented in the FES. The Board finds that the current qualitative assessment of the uncertainties in the FES is adequate.

416. The methodology of WASH-1400 is appropriate for use in preparing a probabilistic assessment of severe accidents for NEPA purposes. (Wohl, NRC Staff Prepared Testimony at 6-8, ff. Tr. 2091; Levine, Applicant Prepared Testimony at 14-15, ff. Tr. 1930.)

417. The Lewis Committee Report (NUREG/CR-0400) supports the application of the WASH-1400 methodology. The authors of the Lewis Report concluded that the probabilistic methodology employed by WASH-1400 is sound and that the event tree/fault tree approach, with an adequate data base, is the best available tool with which to quantify the accident probabilities associated with nuclear power stations. (Wohl, NRC Staff Prepared Testimony at 6-7, ff. Tr. 2091; Levine, Applicant Prepared Testimony at 9-10, ff. Tr. 1930.)

418. The Lewis Committee did criticize some aspects of WASH-1400, but these criticisms did not challenge the study's basic methodology. For instance, the Lewis Report states that the committee was unable to determine whether WASH-1400's overall core melt probabilities were high or low, and concluded that the error bands were understated. The Committee also found it difficult to follow the detailed thread of calculations through WASH-1400. (Wohl, NRC Staff Prepared Testimony at 6, ff. Tr. 2091; Levine, Applicant Prepared Testimony addendum, ff. Tr. 1930.)

419. Although the Commission withdrew its support of the Executive Summary of WASH-1400, the Commission has not withdrawn its endorsement of the study. (Levine, Applicant Prepared Testimony at 15, ff. Tr. 1930.) The Commission recognizes the concerns expressed by the Lewis Report but still supports the use of probabilistic risk assessment. (Wohl, NRC Staff Prepared Testimony at 7-8, ff. Tr. 2091; Levine, Applicant Prepared Testimony at 10-11, 12-13, ff. Tr. 1930; Levine, Tr. 2079-2080.)

420. Other committees and commissions have endorsed the use of WASH-1400's methodology. The reports of the President's Commission on the accident at Three Mile Island and the NRC Special Inquiry Group strongly support the use of PRA in generic regulatory decisionmaking. (Levine, Applicant Prepared Testimony at 11-12, ff. Tr. 1930.)

421. In 1981, NRC Chairman Palladino stated that a Generic Requirement Review Committee should use PRA techniques where sufficient data exists. (Levine, Applicant Prepared Testimony at 12, ff. Tr. 1930.)

422. The Commission's June 13, 1980 Statement of Interim Policy calls for environmental impact statements to include a discussion on the probability radiation could be released to the environment due to an accident and the consequences of such releases. (Wohl, NRC Staff Prepared Testimony at 8, ff. Tr. 2091; Levine, Applicant Prepared Testimony at 12-13, ff. Tr. 1930.)

423. A recent report, presenting the results of a program performed at Oak Ridge National Laboratory, entitled "Precursors to Potential Severe Core Damage Accidents: 1969-1979 A Status Report," (NUREG/CR-2497) ("Precursor Study") estimated the frequency of severe core damage accidents to be between 1.7×10^{-3} and 4.5×10^{-3} per reactor year. WASH-1400 estimated the core melt frequency for the reactors analyzed in the study to be 5×10^{-5} per year. (Newberry, NRC Staff Prepared Testimony at 8-9, ff. Tr. 2091.)

424. The methodology used in the Precusor Study took a rare event at a specific reactor, such as the Browns Ferry fire, and evaluated it using a generic event tree. (Levine, Applicant Prepared Testimony at 25, ff. Tr. 1930; Levine, Tr. 2022-2023.)

425. Using a generic event tree to evaluate a rare event at a specific plant does not take into account the specific event trees or the specific system failure probabilities applicable to that plant. (Levine, Applicant Prepared Testimony at 25, ff. Tr. 1930; Levine, Tr. 2051, 2054-2055.)

426. The Institute for Nuclear Power Operations ("INPO") has applied specific events to plant-specific event trees and obtained estimates much smaller than those in the Precursor Study. (Levine, Applicant Prepared Testimony at 25-26, ff. Tr. 1930; Levine, Tr. 2023, 2028-2032.)

427. The Precursor Study estimates the frequency of severe core damage accidents while WASH-1400 estimated the core

melt frequency. The former category involves damage to the geometric configuration of the core with little release of radiation to the environment because containment integrity is maintained. More serious core melt and containment failure scenarios involving significant releases of radiation to the environment were considered in WASH-1400 because the objective of the study was to estimate the risk to the public from major accidents in nuclear power plants. (Levine, Tr. 2036-2037, 2277-2281.)

428. The Byron population density is generally much less than the average population density of 111 existing or proposed nuclear power reactor sites. (Levine, Applicant Prepared Testimony at 15-16, ff. Tr. 1930.)

429. Based on the evidence, the Board finds that the use of the probabilistic risk assessment methodology of WASH-1400 in preparing Sections 5.9.4.5(2)-(7) and 5.9.4.6 of the FES was appropriate and that the risks to the public from severe accidents at Byron Station has been adequately assessed in the FES. Neither the Lewis Committee Report nor the Precursor Study invalidate the use of the WASH-1400 methodology.

Contention 62: Accident Mitigation

430. Before the NRC will license a nuclear power plant to operate, the applicant must show that it has satisfied numerous NRC regulations and requirements to ensure that the plant's operation will not unduly risk the public's health and safety. These requirements cover both the design and the

construction of the plant. (Levine, Applicant Prepared Testimony at 3-4, ff. Tr. 1930.)

431. Specifically defined accidents, called design basis accidents, are analyzed as a part of the licensing process to demonstrate that significant amounts of radioactivity will not be released from the plant. As a result, all safety systems have redundant components to ensure that the failure of one component will not cause an entire system to fail. (Wohl, NRC Staff Prepared Testimony at 16, ff. Tr. 2091; Levine, Applicant Prepared Testimony at 3-4, ff. Tr. 1930.)

432. The NRC process for making safety evaluations is called "deterministic" because the elements which must be considered in the evaluations are determined by qualitative engineering judgment instead of by quantitative probabilistic estimates. (Levine, Applicant Prepared Testimony at 4, ff. Tr. 1930.) Qualitative engineering judgment is based on the wisdom, the engineering opinion, experience, and the judgment of skilled professionals. (Levine, Tr. 2074-2076.)

433. The NRC's safety evaluation process has been used world wide and has produced nuclear power plants with good safety records. (Levine, Applicant Prepared Testimony at 4-5, ff. Tr. 1930.)

434. The NRC's safety evaluations do not make quantitative estimates of the risk to the public from the plant and no regulation requires that a site-specific PRA be done for plants like Byron. (Wohl, NRC Staff Prepared Testimony at 13, ff. Tr. 2091; Levine, Applicant Prepared Testimony at 5, ff. Tr. 1930; Levine, Tr. 1947.)

435. Although the NRC Staff uses the probabilistic risk assessment approach to assess the environmental impacts from the operation of nuclear power stations, it does not regulate nuclear plant safety on the basis of the conclusions of WASH-1400 or any other probabilistic risk assessment because it is rapidly developing methodology and PRA predictions of public risk still have large uncertainties associated with them. (Wohl, NRC Staff Prepared Testimony at 13, ff. Tr. 2091; Levine, Applicant Prepared Testimony at 3-5, 7, 15, ff. Tr. 1930.)

436. After the accident at Three Mile Island ("TMI"), changes in Byron Station's design and operating features were made specifically to provide protection against the occurrence of a severe accident. (Wohl, NRC Staff Prepared Testimony at 16-20, ff. Tr. 2091; FES, NRC Staff Exhibit 2 at 5-40, §5.9.4.4(1); Levine, Applicant Prepared Testimony at 22, ff. Tr. 1930.)

437. These changes include the installation of a reactor vessel head vent system, the safety parameter display system, core saturation monitors, and reactor vessel water level indicators. Post-TMI modifications also resulted in improved accident monitoring instrumentation and emergency response facilities. (Levine, Applicant Prepared Testimony at 22, ff. Tr. 1930.)

438. The licensee is now required to review the adequacy of many of its procedures and the capabilities of its employees. This reduces the probability of a malfunction or human error by enhancing the maintenance and operation of the

plant's systems and by significantly increasing the ability of the operators and plant management to recognize and respond to any malfunction that does occur. (Wohl, NRC Staff Prepared Testimony at 18-20, ff. Tr. 2091; Levine, Applicant Prepared Testimony at 22, ff. Tr. 1930.)

439. Byron Station had protective features before TMI which were designed to prevent or mitigate severe accidents. (Levine, Applicant Prepared Testimony at 18-22, ff. Tr. 1930.)

440. Byron has the following protective features: a reactor protection system which assists in preventing severe accidents by shutting down the neutron chain eaction; an emergency core cooling system which is designed to prevent severe accidents by preventing reactor core melting if normal fuel cooling water is lost; containment building features which provide protection against design basis and more severe accidents by preventing the release of significant amounts of radiation to the environment; containment sprays and fan coolers which mitigate the consequences of severe and design basis accidents; and an auxiliary feedwater system which is designed to provide an alternative water source to the secondary side of the steam generators if the main feedwater supply is lost. (Levine, Applicant Prepared Testimony at 19-21, ff. Tr. 1930.)

441. Based on the uncontroverted evidence, the Board finds that Byron Station's design provides, consistent with the requirements of the Commission's regulations, adequate protection against severe accidents.

Contention 2a: Incremental Risk From Byron Station

442. The probability of a core melt accident at nuclear power plants is generally about one in ten thousand per reactor-year (Levine, Applicant Prepared Testimony at 22-23, ff. Tr. 1930.)

443. The likelihood of more than one severe accident occurring at more than one plant that results in cumulative significant radiological consequences to the same specific area is significantly smaller than the probability of a severe accident at a single plant. (Wohl, NRC Staff Prepared Testimony at 21-22, ff. Tr. 2091.)

444. The risk of incurring adverse health effects as a consequence of an accident at Byron Station is exceedingly small. The radiation exposures from any accident is roughly comparable to the exposures to both individuals and the general population from normal station operations over the expected lifetime of Byron Station. (Wohl, Branagan, NRC Staff Prepared Testimony at 20, ff. Tr. 2091; FES, NRC Staff Exhibit 2 at 5-66 through 5-67, §5.9.4.6.)

445. The principal health effects that could occur as a result of a nuclear plant accident are early fatalities and latent cancer fatalities. (Levine, Applicant Prepared Testimony at 23, ff. Tr. 1930.)

446. PRA evaluations have shown that the accident risk of an early fatality to people living fifteen miles or more from a nuclear station is extremely small. No increased

risk of early fatalities exists from the coupling of the risks from the plants located in northern Illinois. (Levine, Applicant Prepared Testimony at 23, ff. Tr. 1930; Levine, Tr. 2019.)

447. PRA studies show that the probability of a latent cancer fatality caused by exposure to radiation from a severe nuclear station accident is negligibly small when compared to the probability of an individual dying from cancer contracted from other sources. The number of nuclear reactors located in an area is inconsequential to an individual's cancer fatality risk. (Levine, Applicant Prepared Testimony at 23-24, ff. Tr. 1930; Levine, Tr. 2020.)

448. Based on the uncontroverted evidence, the Board finds that the incremental risk to residents in Northern Illinois, when the risk of accidents at Byron Station is taken into account with the risk from accidents at other nearby nuclear stations, is very small. The possibility of cumulative doses to residents in the northern Illinois area from accidents at more than one nuclear power plant does not endanger the health and safety of the public.

The foregoing document, "Commonwealth Edison Company's Proposed Findings of Fact and Conclusions of Law Regarding Class 9 Accidents" is respectfully submitted by the undersigned attorneys for Commonwealth Edison Company.

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Dated: July 1, 1983

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSICN

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
COMMONWEALTH EDISON COMPANY	)	Docket Nos. 50-454 OL
	)	50-455 OL
(Byron Nuclear Power Station,	)	
Units 1 & 2)	)	

CERTIFICATE OF SERVICE

The undersigned, one of the attorneys for Commonwealth Edison Company, certifies that she filed the original and two copies of the attached "COMMONWEALTH EDISON COMPANY'S PROPOSED FINDINGS OF FACT AND CONCLUSIONS OF LAW REGARDING CLASS 9 ACCIDENTS" with the Secretary of the Nuclear Regulatory Commission and served a copy of the same on each of the persons at the addresses shown on the attached service list. Service on the Secretary and all parties, unless otherwise indicated, was made by deposit in the U.S. Mail, first-class postage prepaid, this 1st day of July, 1983.

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Docket Nos. 50-454 and 50-455

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