UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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| In the Matter of | ) | |
| | ) | |
| COMMONWEALTH EDISON COMPANY | ) | Docket Nos. 50-454 |
| | ) | 50-455 |
| (Byron Station, Units 1 and 2) | ) | |

# NRC STAFF PROPOSED FINDINGS OF FACT AND CONCLUSIONS OF LAW IN THE FORM OF A SUPPLEMENTAL PARTIAL INITIAL DECISION ON CLASS 9 ACCIDENTS AND LIQUID PATHWAY RELEASES

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The NRC Staff, in accordance with 10 CFR § 2.754 and the Licensing Board's directive of May 25, 1983, proposes the following supplemental findings of fact and conclusions of law.

## I. INTRODUCTION

Evidentiary hearings were held in the captioned proceeding from March 1 through May 25, 1983, with some adjournments. There were eight contested issues adjudicated during the period. Proposed findings on five contested issues[1/] have already been filed by the parties. Findings

1/ The Staff has filed its proposed findings on seismology (A-1 through A-103), water hammer (B-1 through B-45), occupational radiation safety (C-1 through C-130), steam generator tube integrity (D-1 through D-243) and emergency planning (E-1 through E-153).

on the Class 9 accident and liquid pathways contentions are filed herewith.[2/] Findings on the final contention regarding quality assurance/quality control are due to be filed on July 20, 1983, although the record has been reopened for further evidence on one aspect of the quality assurance/quality control contention. See "Memorandum and Order Reopening Evidentiary Record," dated June 21, 1983.

## II. OPINION

### F. Class 9 Accidents (League Contentions 8 and 62, DAARE/SAFE Contention 2A)

Three contentions were raised by the Intervenors on the general subject of severe accidents at Byron Station. League Contention 8 asserts that there has been no adequate evaluation of the risks associated with the operation of Byron Station and that there has been no demonstration that the risks are low enough to justify licensing Byron Station to operate. In Contention 62, the League alleges that the design of Byron Station fails to protect the public against that category of severe accidents sometimes referred to as "Class 9" accidents. DAARE/SAFE Contention 2A states that the cumulative risk to residents of northern Illinois from the addition of Byron Station to the

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2/ Class 9 accident findings are designated F-1 through F-106. Liquid pathway release findings are G-1 through G-138.

several other nuclear power reactors in the general area is such that Byron should not be licensed to operate.[3/]

Intervenors offered no affirmative evidence in support of these contentions, relying instead on cross-examination of the witnesses presented by the Applicant and the Staff. For the reasons discussed below, we decide that League Contentions 8 and 62 and DAARE/SAFE Contention 2A are not well-founded. In sum, we find as follows: (1) the Final Environmental Statement ("FES") for Byron Station contains an assessment of the risks associated with the operation of Byron Station which is reasonable and which meets applicable regulatory requirements; (2) the design of Byron Station satisfies regulatory requirements and provides substantial protection against so-called Class 9 accidents; and (3) licensing of Byron Station for operation does not present unacceptable levels of incremental or cumulative risk to the residents of northern Illinois.

1. Regulatory Background

The question of risk enters into the licensing of a nuclear power reactor both in terms of NRC's evaluation of safety and its assessment of environmental impacts associated with reactor operation. Under the

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3/ A related DAARE/SAFE Contention (Contention 2) alleged that the cumulative risk from normal operation (as opposed to risks from accidents) was such that Byron should not be licensed to operate. This contention was resolved against DAARE/SAFE on a motion for summary disposition. See "Memorandum and Order Ruling on Motions for Summary Disposition of DAARE/SAFE Contentions," dated September 10, 1982.

Atomic Energy Act[4/] and the Commission's safety regulations,[5/] a nuclear power reactor may be licensed for operation only after a finding that there is reasonable assurance that no undue risk is presented to the health and safety of the public. See 10 CFR § 50.57(a)(3)(i); Citizens for Safe Power, Inc. v. NRC, 524 F.2d 1291 (D.C. Cir. 1975). The NRC's determination that a particular facility poses no undue risk to public health and safety (including the risk from Class 9 accidents) is based on a deterministic review process rather than on any facility-specific probabilistic assessment of risk.[6/] Under the National Environmental Policy Act ("NEPA"),[7/] the Commission must consider environmental factors in granting, denying or imposing conditions on a license for a nuclear power reactor. Calvert Cliffs Coordinating Committee v. AEC, 449 F.2d 1109 (D.C. Cir. 1971). It is the policy of the Commission to include a reasoned consideration of the environmental risks[8/] attributable to accidents, including very low probability accidents, at a particular facility in its NEPA-required environmental impact statements for the

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4/ 42 U.S.C. § 2239 et seq.

5/ See 10 CFR Chapter 1, particularly 10 CFR § 50.57(a)(3).

6/ See Findings F-6, F-7, F-13 and pages 5-8, infra.

7/ 42 U.S.C. § 4321 et seq.

8/ Environmental risk entails both the probability of a release of radiation and the consequences of such a release. See "Nuclear Power Plant Accident Considerations Under The National Environmental Policy Act of 1969," 45 Fed Reg. 40101, 40103 (June 13, 1980) (hereinafter "Statement of Interim Policy").

facility. Statement of Interim Policy, 45 Fed. Reg. 40101.[9/] Probabilistic consideration of environmental risk is called for by the Statement of Interim Policy. (Finding F-58).

Certain more severe kinds of very low probability accidents are commonly referred to as severe accidents, beyond design basis accidents, or "Class 9" accidents. Such accidents are the focus of Intervenors' Contentions 8, 62 and 2A. The term "Class 9" accident derives from a now-withdrawn Annex to 10 CFR Part 50, Appendix D published for comment in 1971. The Annex explained that the occurrences in Class 9 "involve sequences of postulated successive failures more severe than those postulated for the design basis for protective systems and engineered safety features." See Statement of Interim Policy, 45 Fed. Reg. at 40103; Offshore Power Systems (Floating Nuclear Power Plants), CLI-79-9, 10 NRC 257, 258 (1979). The radiological consequences of such accidents could be, though need not be, severe. (Finding F-85). In accordance with the Statement of Interim Policy, Class 9 accidents are now considered in the Staff's environmental impact statements for power reactor operating license applications.

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9/ NEPA itself does not require that very low probability accidents be considered by the Commission's environmental impact statements. Hodder v. NRC, 589 F.2d 1115 (D.C. Cir. 1978), cert. denied, 444 U.S. 829, rehearing denied, 444 U.S. 974 (1979). While the Commission has directed in its Statement of Interim Policy that an assessment of the environmental risks presented by accidents more severe than the design basis events be presented (thereby opening this area for litigation), we note that the Commission's regulations do not require explicit design consideration for safety purposes of Class 9 accidents. An applicant need not go beyond the requirements of the regulations in order to obtain an operating license. See note 11, infra. Absent a showing of special circumstances to support the invocation of 10 CFR § 2.758, these "severe accident" contentions must and will be treated as environmental rather than safety contentions. As will be seen, however, the evidence demonstrates that no significant safety concerns have been presented and that the ultimate safety finding required by 10 CFR § 50.57(a)(3) is adequately supported.

One important threshold matter can be resolved before proceeding to discuss the evidence presented on these contentions. Intervenor League has suggested that Byron Station cannot be licensed in the absence of a facility-specific probabilistic risk assessment.10/ This suggestion is without merit. A finding of compliance with the regulations entitles one to the requested permit or license insofar as the requirements of the Atomic Energy Act are concerned.11/ No regulation, policy or Staff practice requires that a facility-specific probabilistic risk assessment must be done for Byron Station. (Finding F-14).12/

Twice in recent months the Commission has explained that probabilistic risk assessment is not generally required for operating licensing decisions by regulations or policy. In its "Policy Statement On Safety Goals For The Operation of Nuclear Power Plants," 48 Fed. Reg. 10772, 10775 (March 14, 1983), the Commission discussed the implementation of its proposed safety goals:

> The qualitative safety goals and quantitative design objectives contained in the Commission's Policy Statement will not be used in the licensing process or be interpreted as requiring the performance of probabilistic risk assessments by applicants or licensees during the evaluation period. The goals and objectives are also not to be litigated in the Commission's hearings. The staff should continue

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10/ The League's position on this point has not been consistent. During discovery, the League stated that it was not arguing that a facility-specific probabilistic risk assessment was required. See "Intervenor Rockford League of Women Voters' Answers To NRC Staff Second Set of Interrogatories," dated December 14, 1982, at 1. During hearings, it appeared to be the League's position that such an assessment was required. See, e.g., Tr. 2320-21.

11/ Maine Yankee Atomic Power Company (Maine Yankee Nuclear Power Plant, Unit 2), ALAB-161, 6 AEC 1003 (1973), affirmed, CLI-74-2, 7 AEC 2 (1974), affirmed sub nom., Citizens for Safe Power v. NRC, 524 F.2d 1291 (D.C. Cir. 1975).

12/ Probabilistic risk assessments are required by 10 CFR § 50.34(f) for certain pending construction permit applications. This rule is not applicable to Byron Station.

to use conformance to regulatory requirements as the exclusive licensing basis for plants. (emphasis added).

Similarly, the Commission's "Proposed Commission Policy Statement On Severe Accidents And Related Views On Nuclear Reactor Regulation," 48 Fed. Reg. 16014, 16015-16016 (April 13, 1983), discussed the Commission's proposed policy on the use of probabilistic risk assessment in severe accident decisionmaking. The proposed policy statement emphasized the primacy of "the current deterministic approach for reviewing design and operation of a nuclear power plant." While special circumstances have caused the NRC to require that probabilistic risk assessments be done at two operating reactors (Indian Point and Zion),[13/] no such action has been taken by the Commission for Byron Station.

We conclude from the lack of any clear affirmative requirement based in regulation, policy or practice, from the Commission's cautionary statements just cited, and from the limited special cases where probabilistic risk assessment does enter into the licensing process that a facility-specific probabilistic risk assessment is not a requirement for the licensing for operation of Byron Station. Moreover, as discussed in detail below, the League has failed to show that the discussion of severe accidents in the FES for Byron Station is inadequate in the

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13/ In the cases of both Indian Point and Zion, high population densities near the site caused the NRC to require probabilistic risk assessments. (Finding F-15). In one operating license proceeding (Limerick), the Staff requested a probabilistic risk assessment of the applicant for the same reason. See Philadelphia Electric Co. (Limerick Generating Station, Units 1 and 2), LBP-82-43A, 15 NRC 1423, 1489-91 (1982). Byron Station is not at a high population density site (Finding F-16) and no probabilistic risk assessment has been requested or required for Byron Station. Neither is one needed to demonstrate compliance with the regulations.

absence of a facility-specific probabilistic risk assessment. The League is, thus, attempting to impose on Edison a requirement which does not appear in the Commission's regulations. This can only be done by recourse to 10 CFR § 2.758. See Pacific Gas and Electric Co. (Diablo Canyon Nuclear Power Plant, Units 1 and 2), ALAB-723, 17 NRC ____ (May 18, 1983) (slip op. at 61). No attempt has been made by Intervenors to pursue their claim that such an assessment is needed by a motion pursuant to 10 CFR § 2.758, nor do we believe on the entire record of this proceeding that the requisite showing of special circumstances for an exception to the Commission's rules and regulations can be made.

Having resolved this threshold legal issue, we turn now to consider the specific contentions raised by the Intervenors.

2. League Contention 8

Contention 8 alleges that there has been no "meaningful assessment of the risks associated with the operation of the proposed Byron nuclear facility, contrary to the requirements of 10 CFR § 51.20(a) and § 51.20(d)." The contention goes on to state that the potential for severe accidents exists, that the Reactor Safety Study (WASH-1400) should not be relied upon in assessing risk, that consideration has not been given to Class 9 accidents, and that there is "no technical basis for concluding that the actual risk is low enough to justify operation of Byron." (Finding F-1).

There is no dispute among the parties that the potential for accidents at Byron Station, including severe accidents, exists. Contrary to the League's contention, however, a lengthy and detailed discussion of the risks posed by accidents at Byron and the impacts of such accidents is contained in the FES. (Finding F-18). The FES considers a number of postulated accidents of several types ranked on the basis of their relative likelihood. (Finding F-19). It also considers accidents of greater severity -- the Class 9 accidents. (Findings F-20, F-21). Attention is given both to the probability of occurrence of radioactive releases and to the probability of occurrence of the environmental consequences of those releases via atmospheric and groundwater pathways. (Finding F-29). The conclusion is drawn in the FES that the potential environmental impacts from accidents at the Byron Station could be severe but that the likelihood of their occurrence is small. (Finding F-28). The overall assessment of environmental risk of accidents, assuming protective action, shows that it is roughly comparable to the risk from normal operation although accidents have a potential for early fatalities and economic costs that cannot arise from normal operations. (Finding F-28). In numerical terms, the average value for early fatalities resulting from a reactor-year of operation at Byron is estimated at 0.00026, which is an extremely small fraction of the total risk of accidental fatality in the United States. (Finding F-26).

The probabilistic risk assessment methodology pioneered in WASH-1400 clearly is used by the Staff in the preparation of its FES discussion of risks. (Finding F-33). Indeed, the Statement of Interim Policy mandates use of probabilistic risk assessment methodology. (Finding F-58). Contrary to the League's contention, that methodology has not been called into question since publication of WASH-1400. (Finding F-35). The Lewis Report, cited by Contention 8, states that the fault-tree/

event-tree methodology used in WASH-1400, coupled with an adequate data base, is the best available tool with which to quantify the accident probabilities associated with nuclear reactors. (Finding F-36).[14/] As used in the FES, WASH-1400 has been updated, or "rebaselined," to incorporate peer review group comments and better data and analytical techniques now available. (Finding F-22).

Edison and the Staff addressed the impact on WASH-1400 and its methodology of a recent report known as the "Precursor Study."[15/] The Precursor Study used operational data in Licensee Event Reports to evaluate potential accident precursors occurring at operating reactors and to derive a probability for severe core damage. (Findings F-38, F-39). The probability derived by the Precursor Study is approximately two orders of magnitude greater than that estimated in WASH-1400. (Finding F-39).

Both Edison's expert, Saul Levine, and the Staff's witnesses concluded that the Precursor Study probability estimates are too high

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14/ As Contention 8 notes, the Commission has stated that it does not regard as reliable the numerical estimate of overall risk provided by WASH-1400. Nevertheless, the Commission supported the extended use of probabilistic risk assessment in regulatory decisionmaking, taking due account of the reservations expressed in the Lewis Report. See "NRC Statement On Risk Assessment And The Reactor Safety Study (WASH-1400) In Light Of The Risk Assessment Review Group," (January 18, 1979); Petition For Emergency And Remedial Action, CLI-80-21, 11 NRC 707, 722 n.23 (1980). It is the methodology of WASH-1400, rather than its numerical estimates of overall risk, that is used by the Staff in its FES for Byron Station.

15/ "Precursors to Potential Severe Core Damage Accidents; 1969-1979, A Status Report" (NUREG/CR-2497).

and do not invalidate the estimates of WASH-1400 or the use of its methodology. (Findings F-40, F-50). Mr. Levine emphasized that simplified models and simplified assumptions had been used in the Precursor Study which yielded failure probabilities that are too high and that are not directly comparable to WASH-1400's estimates. (Findings F-51 through F-54). The Staff noted that a substantial fraction (82%) of the Precursor Study's estimate of severe accident frequency comes from events for which subsequent ameliorating or corrective actions have been taken. (Findings F-41 through F-48). We conclude that the Precursor Study does not invalidate either the probability estimates or the methodology of WASH-1400 and does not call into question the Staff's reliance on WASH-1400, as rebaselined, in the FES.

Conflicting evidence was presented on one issue related to this contention -- the issue of uncertainty in the FES estimates. The Statement of Interim Policy requires a discussion of "the extent to which events arising from causes external to the plant are considered possible contributors to the risks associated with the particular plant;" it also states that major uncertainties should be identified. (Findings F-57, F-59).

The parties agree that uncertainty enters into probabilistic risk assessment in several ways, including the frequencies of external events. (Finding F-60, F-61). The FES considered external events only qualitatively; good quantification of the risks from external events (including sabotage) is generally considered beyond the state of the art. (Findings F-63, F-64, F-76). In its testimony, however, the Staff drew upon information from studies at Indian Point and Zion which indicated that external events could be significant contributors to risk and assigned an uncertainty factor of 100 to attempt to quantify the upper bound of accident risks from internal and external causes. (Findings F-70 through F-76).

Edison filed rebuttal testimony through Mr. Levine addressing the Staff's uncertainty factor of 100. Mr. Levine agreed that the actual probabilities associated with the operation of Byron are within the bounds of the factor of 100 but did not agree that the range of uncertainty should be as large as a factor of 100 in the direction of increased risk. (Finding F-78). Mr. Levine objected to the use of the results of probabilistic risk assessments for other plants as the basis for establishing quantitative uncertainty factors to be applied to the Byron FES estimates. (Finding F-79). He also questioned whether it was necessary or useful to provide uncertainty estimates on the risk results in the FES, given the generalized and, in his view, conservative approach taken there. (Finding F-80).

There is a large amount of judgment involved in estimating uncertainties. (Findings F-60 through F-62). This highly judgmental issue need not be resolved by this Board in order to arrive at a decision concerning the licensing for operation of Byron Station. The Commission's rules do not require that quantitative estimates of uncertainty be provided. (Finding F-81). The Staff has, nevertheless, attempted to provide quantitative bounds to the uncertainty in the FES. (Findings F-71, F-72, F-75, F-76). While we find the Staff's collective judgment on the uncertainty issue reasonable under the circumstances, the pivotal point for the purpose of this contention is one on which the testimony is in accord: the FES estimation of risk for Byron is reasonable and is responsive to the requirements imposed by the Statement of Interim Policy. (Findings F-29, F-32, F-77, F-82, F-83).

In sum, it is the current deterministic approach of evaluating conformance to regulatory requirements, and not probabilistic risk assessment, which provides the basis for safety licensing decisions. As to the assessment of environmental risks associated with the operation of Byron Station, the FES estimation and discussion of risk is reasonable and meets the requirements of the Commission's Statement of Interim Policy.

3. League Contention 62

Contention 62 asserts simply that the design of Byron does not provide protection against Class 9 accidents and that, for this reason, there is no reasonable assurance that Byron can be operated without endangering the health and safety of the public. (Finding F-1). Any environmental aspects of this contention are subsumed by our discussion of Contention 8 above. With respect to the safety issue raised, while we believe that Contention 62 is an impermissible challenge to the Commission's regulations for the reason discussed in note 9, supra, we nevertheless find that the design of Byron provides substantial protection against Class 9 accidents.

Each nuclear power plant licensed by the NRC must be shown to meet an extensive set of deterministic requirements to provide reasonable assurance that operation of the plant will not present undue risk to the health and safety of the public. (Finding F-6). A reactor and its various safety systems are analytically tested for adequacy of performance against a series of design basis events. (Finding F-6, F-9). The conservative assumptions used in these analyses result in a design capability with multiple and redundant systems for coping with very severe performance demands and with substantial protection against unforeseen events and severe accidents. (Findings F-89, F-90). Emergency operating procedures have been prepared based on various event sequences (including multiple failures) beyond the design basis events. (Finding F-91). Moreover, many additional requirements have been imposed since the TMI accident to reduce the likelihood of Class 9 accidents. (Findings F-94, F-95, F-96).

We find that the deterministic licensing requirements, knowledge acquired from the TMI accident, mitigative engineered safety features, multiple barriers against post-accident release of radioactivity and additional measures such as emergency operating guidelines which allow risk-reducing human intervention in reactor accident situations, provide reasonable assurance that the Byron plant can be operated with no undue risk to the public health and safety. (Finding F-97).

4. DAARE/SAFE Contention 2A

DAARE/SAFE alleges in Contention 2A that the addition of Byron Station creates the potential for cumulative doses to residents in northern Illinois from discrete accident events at the several plants in the general area and thereby raises to an unreasonable level the risk to the health and safety of area residents. The evidence does not support this contention.

There has been no measured radiological dose burden to northern Illinois residents from nuclear power plants operating in the area. (Finding F-99). The issue of the risk associated with placing a number of reactors within a certain geographic area has been studied, particularly in the context of multiple reactor siting. (Finding

F-102). At the distances involved here,[16/] the cumulative risk to area residents from accidents at Byron Station and at other plants in the northern Illinois area is negligible. (Findings F-103 through F-106). In particular, the likelihood of a severe accident occurring at any of the nuclear power plants in northern Illinois is sufficiently small that the addition of the Byron reactors will not raise this likelihood to a significant level.[17/] (Finding F-100).

We find that the possibility of cumulative doses to residents of the northern Illinois area from accidents at more than one nuclear power plant does not create undue risk to public health and safety. (Finding F-98).

5. Conclusion on Class 9 Accident Contentions

We conclude that League Contentions 8 and 62 and DAARE/SAFE Contention 2A lack merit. The FES for Byron Station contains an assessment of the risks associated with the operation of Byron Station

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16/ Rockford is located 15 miles from the Byron Station and 60 miles or more from Zion Station, the next nearest plant to Rockford. DeKalb and Sycamore are about 30 miles from the Byron Station and about 40 miles or more from the Dresden and LaSalle stations, which are about equally distant from the communities. (Finding F-104). The accident risk of early fatality to people living at distances of 15 miles or more from a nuclear plant is exceedingly small. (Finding F-103). As to latent cancers, the probability is very small whether there is one reactor or ten reactors in the area. (Finding F-106).

17/ In our September 10, 1982 "Memorandum and Order Ruling on Motions for Summary Disposition of DAARE/SAFE Contentions," we raised a concern about the effect of the then-recent Precursor Study on the evidence proffered by the movants for summary disposition. The evidence at hearing has considered the Precursor Study and shows that the Precursor Study results do not cause a change in the population dose estimates made in the FES. (Finding F-49).

which is reasonable and which meets applicable regulatory requirements. The design of Byron Station satisfies regulatory requirements and provides substantial protection against so-called Class 9 accidents. Finally, the licensing of Byron Station for operation does not present unacceptable levels of incremental or cumulative risk to the residents of northern Illinois.

## G. Liquid Pathway Releases (League Contentions 39 and 109, as Consolidated)

Two of the League's contentions dealt with the specific subject of liquid pathway releases of radionuclides to the environment.[18/] (*See* Findings G-1, G-2). The liquid pathways are routes by which people can be exposed to radiation released by a nuclear power plant via surface water and groundwater. Exposure, for example, may be from drinking or swimming in contaminated water. (Finding G-10). These liquid pathway contentions were consolidated and rephrased by stipulation. (Finding G-3). The consolidated contention alleges that the groundwater system underlying the Byron Station site has not been characterized adequately, that the consequences of a radioactive release to the aquifer cannot be predicted with confidence, and that neither NEPA nor the appropriate safety regulations are satisfied here. (Finding G-3).

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18/ A liquid pathway release caused by a core melt accident is a narrower aspect of the Class 9 accident issue discussed in Section F of this Supplemental Partial Initial Decision. The legal basis for an evaluation of the consequences of a liquid pathway release is the same as discussed for severe accidents generally at pages 3-8 of this decision.

Intervenor League, Edison and the Staff each presented expert witnesses to address the liquid pathways consolidated contention. (Findings G-4, G-5, G-6). The League's expert, Dr. Wood, explained the League's position that the extensive fracturing of the dolomite limestone bedrock beneath the site creates a particular hazard in the event of a release of radionuclides which has not been analyzed properly by the Applicant or the Staff. Dr. Wood suggested that a more site-specific model of the aquifer should be developed, that more groundwater movement data should be obtained and that a more detailed analysis should be conducted. (Findings G-82 through G-87). Having considered all of the evidence presented on this contention, the Board finds that the groundwater system underlying the Byron Station site has been characterized adequately, that the consequences of radionuclide releases via the liquid pathway have been adequately and conservatively addressed in the FES and that a more detailed liquid pathway analysis is unwarranted. (See Findings G-137, G-138). We review the evidence that leads us to these conclusions below.

1. Site Characterization

The testimony on the geological characterization of the site area was extensive, detailed and largely uncontroverted. Applicant's site investigation has shown that the site area is covered with a mantle of glacial drift which is underlain by dolomites and limestones of the Ordovician-age Galena and Platteville Groups. (Findings G-14, G-15). The Galena-Platteville bedrock, which contains groundwater, is perched on the Harmony Hill Shale member of the Glenwood formation. (Finding G-18). This formation acts as a hydraulic barrier preventing contamination of the lower aquifers, which supply almost all drinking water supplies for municipal users in the area. (Finding G-22). Thus, groundwater contamination that might emanate from the Byron Station would travel only through the glacial drift and upper formations of the Galena-Platteville dolomites. (Finding G-23).

The Dunleith Formation within the Galena-Platteville is the upper bedrock unit at the site and serves as the foundation for safety-related structures at the Byron Station. (Finding G-24). This bedrock is fractured, jointed and thin-bedded, but there are no large openings along joints and bedding planes. (Finding G-25). The most highly fractured bedrock occurs in the first 15 to 20 feet; this material has been excavated out at the Byron site. The amount of fracturing decreases dramatically with depth and the bedrock is only moderately to slightly weathered at the point where the Byron Station foundation actually sits. (Finding G-26). In addition, the water table at the Byron Station site is below the most weathered zone of the dolomite. (Finding G-16).

Because of the fractured and jointed characteristics of dolomite bedrock, Edison used pressure rock cement grouting to fill and seal all major enlarged joints, bedding planes and other planar features of the bedrock. (Finding G-32). While the grouting was done primarily to ensure the structural integrity of the plant foundation, it has the effect of significantly reducing the permeability of the rock and, thus, retarding the flow of any accidentally-released effluents. (Findings G-34 through G-37).

2. Analysis of Consequences of Liquid Pathway Releases

The consequences of a release of radionuclides via the liquid pathway have been analyzed by the Staff and the results of that analysis are presented in the FES. (Finding G-38). The Staff takes as its starting point the detailed analysis of liquid pathway release consequences conducted in the "Liquid Pathway Generic Study" (or "LPGS"). The LPGS evaluated the risk of accidents involving the liquid pathway for four conventional, generic land-based nuclear plants; it concluded that the relative contribution to risk of a liquid pathway release is not significant.[19/] (Findings G-39, G-42). The purpose of the liquid pathway evaluation in the Byron FES is to confirm that there is no unique hazard associated with the Byron Station in comparison to the LPGS. If the risk is found not to be significantly greater for Byron Station than for the LPGS case, there is no need to conduct a more detailed analysis. (Findings G-42 through G-44, G-50).

The accident scenario most closely studied by the Staff is the core melt accident with penetration of the basemat, since such accidents are the principal contributors to risk from a liquid pathway release. (Finding G-45). There are two principal mechanisms by which liquid pathway releases could result from a core melt accident. A basemat penetration could release molten core debris to the bedrock beneath the

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19/ The airborne exposure pathway normally provides the greater contribution to risk in an accidental release of radionuclides. The reasons for this include the rapidity of dispersion through the atmosphere, the greater difficulty of avoiding atmospheric exposure, the ability to monitor and interdict liquid pathway releases, and the greater likelihood of acute health effects from exposure to atmospheric releases. (Findings G-40, G-132, G-133).

containment structure. After about a year, during which cooling of the core debris occurs sufficiently to permit groundwater to come into contact with the core debris, a leaching process could result in transport of radionuclides into the groundwater system. Less likely (but more severe in consequence), the basemat penetration could release a quantity of highly contaminated "sumpwater" directly into the groundwater.[20/] Both cases were considered by the Staff. (Findings G-73).

The Staff evaluated the direction and travel time of the postulated releases in assessing the consequences of a core melt accident via the liquid pathway. (Finding G-46). Taking into account doses through drinking water, fish ingestion and shoreline exposure, the Staff concludes that the consequences of a liquid pathway release at Byron Station would be about a factor of three higher than those for the LPGS small river site. (Finding G-47). This is a conservative estimate since

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20/ Edison witness Klopp testified that recent analyses show that the sumpwater source term can be eliminated. (Finding G-74). He reasons that basemat penetration can occur only if no cooling water is present to quench the core; without water, no sumpwater source term exists. (Findings G-75, G-76). He concludes that the FES is "very significantly conservative" because it includes a sumpwater source term. (Finding G-74). The evidence shows that there is some likelihood, however small, that debris bed coolability could be prevented even with water available. (Findings G-78 through G-80). Moreover, even in the absence of a liquid sumpwater source term, basemat penetration could cause the release of a substantial volume of radionuclide-bearing steam (created by the boiling-off of between 100,000 and 400,000 gallons of sumpwater) from the containment atmosphere into the groundwater. (Findings G-77, G-81). Mr. Klopp had not done any analysis of the existence or magnitude of such a source term; it is considered, however, within the FES analysis. (Finding G-81). In light of our overall conclusions concerning the adequacy of the FES treatment of Class 9 accidents generally and the liquid pathway analysis specifically, we find that there is no need to resolve the issues presented by Mr. Klopp's apparent disagreement with certain aspects of the Staff's environmental assessment.

it does not take credit for a number of likely conservatisms, including the likelihood of successful mitigative actions such as interdiction of a liquid pathway release by well-point dewatering or by injection of an impermeable grout curtain. (Findings G-46, G-47, G-48). Because the Byron site liquid pathway contribution to population dose is of the same order of magnitude as that predicted for the comparable LPGS site, the Staff has concluded that the Byron site is not unique and that no more rigorous analysis is necessary. (Findings G-49, G-50).

Intervenor's expert, Dr. Wood, criticized the evaluation presented in the FES on several grounds. Dr. Wood's principal point was that the fractured and jointed bedrock present beneath the Byron Station requires a more detailed analysis than has been done in the FES. He cited the behavior of chemical contaminants at the nearby Byron salvage yard to support his view that a more detailed analysis of the effects of radionuclide releases to groundwater is needed. This analysis would integrate field measurements in a specific model rather than an assumption that a porous medium could adequately describe groundwater travel. (Findings G-82 through G-88).

We find that Dr. Wood's criticisms of the FES evaluation of liquid pathway release consequences are not well-founded. The primary evidence relied upon by Dr. Wood was the cyanide contamination data from the salvage yard. Dr. Wood calculated an extremely rapid groundwater travel time (8 feet per day, as opposed to the 0.52 feet per day indicated by Applicant's investigations at the Byron Station site) from the data on the assumption that the groundwater provided the operative transport mechanism. (Findings G-103, G-104, G-54). This assumption is not established by the data. Indeed, the preponderance of the available data supports the view that the cyanide was carried in significant part by surfacewater routes.[21/] Dr. Wood conceded that one could infer that the surfacewater system was also involved in transporting the cyanide and that his groundwater velocity estimates would be invalidated if surfacewater were the operative or dominant transport mechanism at the salvage yard site. (Findings G-111, G-113). On the record before us, we find this to be the case.

Dr. Wood's velocity estimate of 8 feet per day was calculated simply by dividing the estimated distance traveled by the number of days between deposition and observation downgradient. (Finding G-103). Even apart from the unsupported assumption that transport was via the groundwater, we find that such a calculation has little value when the timing and location of initial deposition are unknown. (Finding G-107 through G-110).

Additional reasons exist for rejecting direct application of the salvage yard data to the Byron Station site. The bedrock beneath the salvage yard site is more fractured near the surface than at the Byron Station site. (Finding G-119). In a core melt accident, the source of the effluent would enter at a lower stratigraphic level where fracturing is significantly less. (Findings G-121, G-26). The effect of the grouting beneath Byron Station must also be taken into account in distinguishing the situation at the salvage yard. (See Findings G-32 through G-37, G-92).

Contrary to Dr. Wood's suggestion, there is no evidence of significant continuous fracturing or jointing below the containment basemat level

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21/ Surfacewater transport of the cyanide is strongly indicated by the death of livestock from contaminated drinking water, by surface drainage features, and by the dumping practices apparently utilized. (Findings G-107 through G-109, G-114 through G-116).

that would lead directly to a surfacewater pathway. (Finding G-92, G-94). Examination of the bedrock and of the location, orientation and approximated aperture sizes of the various joints supports the conclusion that the use of an assumption that the bedrock behaves similarly to a porous medium is appropriate. (Finding G-122, G-123). Testing of the sort advocated by Dr. Wood is difficult and extremely time-consuming[22/] and their use is not warranted here; sufficient reliable information has been accumulated to support the FES evaluation of liquid pathway release consequences. (Findings G-124, G-125, G-135).

We conclude that the groundwater system underlying the Byron site has been characterized adequately, that the Byron site is not unique in its liquid pathway contribution to risk and that there would be ample time to intercept accidental groundwater releases before they could contaminate surfacewater or downgradient wells. (Finding G-137). We further conclude that the consequences of radionuclide releases via the liquid pathway have been adequately and conservatively addressed in the FES and a more detailed liquid pathway analysis is unwarranted. (Finding G-138).[23/]

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22/ Tracer tests, for example, could be used to determine travel time unequivocally. However, if the travel time estimate of approximately 30 years is correct, then tracer tests would themselves take 30 years to complete. (Finding G-124).

23/ Our discussion of the consolidated liquid pathways contention has focused on the environmental assessment of risks caused by liquid pathway releases, but the contention also mentions certain safety regulations--10 CFR §§ 50.37(a)(3)(i), 50.57(a)(6) and 50.34(b)(4). (We disregard the last reference as adding nothing to the previous two). For the reasons discussed at pages 3-8, and particularly note 9, above, the only cognizable safety issue raised by this contention relates to the design basis analysis of a contaminated water tank rupture. Our Findings G-62 through G-65 lead us to conclude that there is reasonable assurance that such an event would not pose undue risk to the health and safety of the public.

## III. FINDINGS

### F. Class 9 Accidents

#### 1. Matter in Controversy

F-1. Three contentions were admitted for litigation on the general subject of severe accidents at Byron Station. League Contention 8 states as follows:

> 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 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 CFR § 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 demonsrate 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 Bryon.

League Contention 62 states:

> 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 no reasonable assurance that Byron can be operated without endangering the health and safety of the public. See also Contention 8, supra.

Finally, DAARE/SAFE Contention 2A is as follows:

> 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 releases on DeKalb-Sycamore and Rockford area reasidents. 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.

F-2. Five witnesses gave testimony on these contentions. For the Applicant, Saul Levine appeared and testified; Mr. Levine is a vice-president and group executive of the consulting group of NUS Corporation. (Tr. 1929 (Levine)). He has considerable knowledge and experience on the subject of nuclear reactor accidents and the probabilistic assessment of risk of such accidents and attendant consequences. (See Levine testimony, ff. Tr. 1930, at 1-2). Mr. Levine was Project Staff Director while at the AEC for the Reactor Safety Study. In this capacity, he was closely involved in the preparation of WASH-1400. (Tr. 1942 (Levine)).

F-3. Four members of the Staff gave testimony. L.G. Hulman is Branch Chief of the Accident Evaluation Branch, Office of Nuclear Reactor Regulation ("NRR"). Millard L. Wohl is a nuclear engineer in that branch. Scott Newberry is a risk analyst in the Reliability and Risk Assessment Branch, NRR. Dr. Edward F. Branagan, Jr., is a health physicist in the Radiological Assessment Branch, NRR; Dr. Branagan had appeared previously as a witness on the ALARA contentions. (See Hulman et al. testimony, ff. Tr. 2091).

F-4. Intervenors did not present any affirmative evidence.

## 2. Regulatory Background

F-5. The regulatory requirements with respect to consideration of severe accidents at nuclear power reactors are provided primarily by a Commission Statement of Policy dated June 13, 1980. 45 Fed. Reg. 40101 (June 13, 1980). (See Hulman et al., ff. Tr. 2091, Attachment B).

F-6. Each nuclear power plant licensed by the NRC must demonstrate that it meets an extensive set of regulatory requirements to ensure that operation of the plant will not present undue risk to the health and safety of the public. In addition, several explicitly defined accidents (called design basis events or accidents) are analyzed in the licensing process to demonstrate that people living near the plant will not be subjected to undue risk from radioactive releases. (Levine testimony, ff. Tr. 1930, at 4).

F-7. If an applicant meets the appropriate regulations, Standard Review Plan criteria and regulatory guides dealing with the particular subject, then adequate safety has been provided. (Tr. 2105 (Levine)).

F-8. The NRC licensing process is deterministic rather than probabilistic. Qualitative engineering judgment has been applied over the years to define those elements that must be considered in safety evaluations. (Levine testimony, ff. Tr. 1930, t 4; Tr. 2304 (Levine)).

F-9. Probabilistic risk assessment ("PRA") is a combination of logic structures (event trees and fault trees) that permits estimates to be made of the likelihood and consequences of accidents that have not been observed because of their low frequency of occurrence. (Levine testimony, ff. Tr. 1930, at 5).

F-10. When the probabilities of various accident sequences have been determined, the physical processes that could occur during these sequences must be analyzed to estimate the amount of radioactivity that could be released to the environment by the various accident sequences. With the probability of releases of various amounts of radioactivity in hand, a further analysis is needed to predict the dispersion of radioactivity in the environment and the health effects induced in people who may be exposed to this radioactivity. (Levine testimony, ff. Tr. 1930, at 6).

F-11. The predictions of public risk in PRAs have large uncertainties which make the use of such predictions in the safety-related licensing of reactors questionable at this time. (Levine testimony, ff. Tr. 1930, at 7).

F-12. While the use of full PRAs is now of little utility in the safety-related licensing process, part of the overall PRA methodology, especially that associated with the prediction of system reliability, can sometimes be of help in resolving safety issues in individual licensing cases. (Levine testimony, ff. Tr. 1930, at 7; Tr. 1947 (Levine)).

F-13. Probabilistic risk assessment does not provide the basis for decisions concerning safety in the licensing of Byron Station. Licensing considerations have rested, and continue to rest, upon an applicant's compliance with the Commission's deterministic licensing criteria. (Hulman _et al_. testimony, ff. Tr. 2091, at 12-13; Levine testimony, ff. Tr. 1930, at 15).

F-14. Probabilistic risk assessments are not generally required by NRC regulations, policy or staff practice. (Levine testimony, ff. Tr. 1930, at 5; Tr. 2066 (Levine); Tr. 2144 (Hulman)).

F-15. PRA's for Indian Point and Zion were required by the NRC because of the relatively high population density at those sites. (Tr. 2115 (Hulman)).

F-16. Byron Station is not located at a high population density site. (Levine testimony, ff. Tr. 1930, at 15-16).

F-17. Performance of a plant-specific PRA is not a licensing requirement for Byron Station. (Hulman _et al_., ff. Tr. 2091, at 13; Tr. 2006 (Levine)).

3. Substantive findings

a. Assessment of environmental risks from operation of Byron Station

(1) Final Environmental Statement assessment of risk

F-18. The Final Environmental Statement for Byron Station, at Section 5.9.4.5, contains a lengthy and detailed discussion of the risks posed by accidents at Byron and the impacts of such accidents. (Staff Ex. 2, at 5-42 through 5-66).

F-19. The FES discussion includes consideration of a number of postulated accidents of several types (called "design-basis" accidents) ranked on the basis of their relative likelihood. Population exposures for these accidents have been calculated and the Staff concludes that "any adverse health effects as a consequence of these events is exceedingly small." The Staff further concludes that "radiation exposures from design-basis accidents are roughly comparable to the exposures to individuals and the population from normal station operations over the expected lifetime of the plant." (Staff Ex. 2, at 5-42 to 5-43; Hulman et al. testimony, ff. Tr. 2091, at 20). (This conclusion is based on a comparison of dose and does not take design basis accident probabilities into account.) (Tr. 2306-07 (Hulman)).

F-20. The FES also includes consideration of accidents of greater severity than the design-basis accidents. The FES explains that severe accidents are distinguishable from design-basis accidents in two ways: (1) they involve substantial physicial deterioration of the fuel in the reactor core, including overheating to the point of melting; and (2) they involve deterioration of the capability of the containment structure to perform its intended function of limiting the release of radioactive materials to the environment. (Staff Ex. 2, at 5-44).

F-21. Sections 5.9.4.5 (2), (3), (4) and (5) discuss the probabilities and consequences of these severe accidents. (Staff Ex. 2, at 5-44 through 5-59).

F-22. The FES utilizes the probabilistic risk assessment methodology described in the Reactor Safety Study (WASH-1400), published in 1975. WASH-1400 quantified the risk associated with operation of large light water reactors; it concluded that the risks are very small compared to the risks to which people are normally exposed. (Tr. 1993 (Levine)). Certain updating, or "rebaselining," has been undertaken since WASH-1400 to incorporate peer group comments and better data and analytical techniques now available. (Staff Ex. 2, at 5-44).

F-23. FES Table 5.11 provides a summary of the probabilities of the rebaselined accident sequences found by WASH-1400 to be the dominant contributors to risk in the prototype PWR. The probabilities and consequences of these accident sequences give a perspective of the societal risk at Byron. (Staff Ex. 2, at 5-44 to 5-45; Tr. 2283 (Wohl)).

F-24. The results of the calculations of dose and health impacts performed for a Byron unit and the site are presented in a series of probability distribution graphs in the FES. (Staff Ex. 2, at 5-48 and Figures 5.6 through 5.9).

F-25. The environmental consequences estimation in the FES takes into account significant site-specific features such as sector-dependent population, meteorology, land fraction data surrounding the site, and liquid pathway consequences. (Hulman et al. testimony, ff. Tr. 2091, at 8; Tr. 2178 (Wohl), 2184 (Hulman)).

F-26. FES Table 5.13 shows average values of risk from Byron operation associated with population dose, acute fatalities, latent fatalities, and costs for evacuation and other protective actions. These probabilities and averages are presented on a per-reactor-year basis. The average value for early fatalities resulting from a reactor-year of operation of Byron is estimated at 0.00026. This is an extremely small fraction of the total risk of accidental fatality in the United States from such causes as motor vehicle accidents, falls, drowning, burns or firearms. The average value for latent cancer fatalities (for all organs, excluding thyroid) is 0.0125 per reactor-year of operation. (Staff Ex. 2, at 5-59 to 5-60; Hulman et al. testimony, ff. Tr. 2091, at 20-21; see Levine testimony, ff. Tr. 1930, at 18).

F-27. The FES discussion of accident risk at Byron and the Staff's testimony herein identify the major sources of uncertainty in such risk estimates. (Staff Ex. 2 at 5-44 to 5-45, 5-65 to 5-66; Hulman et al. testimony, ff. Tr. 2091, at 3-6).

F-28. The Staff concludes in its FES that the potential environmental impacts from accidents at the Byron Station could be severe but that the likelihood of their occurrence is small. The overall assessment of environmental risk of accidents, assuming protective action, shows that it is roughly comparable to the risk from normal operation although accidents have a potential for acute facilities and economic costs that cannot arise from normal operations. (Hulman et al. testimony, ff. Tr. 2091, at 3, 20; Staff Ex. 2, at 5-66 to 5-67).

F-29. The FES contains a reasoned consideration of environmental risks from Byron Station, including risks resulting from postulated accidents. Attention is given there both to the probability of occurrence of radioactive releases and to the probability of occurrence of the environmental consequences of those releases via atmospheric and groundwater[24/] pathways. (Hulman et al. testimony, ff. Tr. 2091, at 2).

F-30. Edison witness Levine testified that the Staff's evaluation of the probabilities and consequences of severe accidents at the Byron Station, as presented in the FES, represents a reasonable and, in his view, conservative prediction of public risks. (Levine testimony, ff. Tr. 1930, at 2-3, 18; Levine rebuttal testimony, ff. Tr. 6956, at 1; Tr. 1949, 2065 (Levine)).

F-31. While a plant-specific PRA for Byron might be helpful in arriving at a judgment as to the estimates in the FES (Tr. 2164 (Newberry)), Mr. Levine testifed that such an assessment would likely show the risks at Byron to be much smaller than those portrayed in the FES (Tr. 6978-79 (Levine)).

F-32. The Staff's estimate of risk is adequate for FES purposes and adequate to meet the requirements of the Commission's Statement of Interim Policy. (Tr. 6991 (Levine)).

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24/ Staff witness Hulman stated during oral testimony that Section 5.9.4.5(5) of the FES would need to be reassessed in light of the indication by Edison that certain liquid pathway recalculations would be required because of an error in the FSAR. (Tr. 2093-94 (Hulman)). That reassessment was done (see Findings G-53 through G-56, infra) and no change to the FES discussion in question was required. (See Codell and Staley testimony, ff. Tr. 6549, at 12).

(2) Adequacy of the methodology utilized by the Staff

F-33. The probabilistic risk assessment methodology of WASH-1400 was used by the Staff in the preparation of Section 5.9.4 of the FES. Probabilistic discussion of environmental risks attributable to accidents at nuclear power reactor facilities is called for by the Commission's June 13, 1980 Statement of Interim Policy. (Hulman et al. ff. Tr. 2091, at 6, 8).

F-34. The charter of the Reactor Safety Study (WASH-1400) was to make quantitative predictions of the risks to the public from potential accidents from 100 operating nuclear power plants. This was done by analyzing in great detail two specific reactors (a pressurized water reactor and a boiling water reactor) and extrapolating this information to an assumed population of 100 reactors at a "composite" site that included the significant characteristics of the sites at which these reactors were located. The site characteristics included population and meteorological features of 68 different sites. The major result of the Reactor Safety Study was that the risk from a population of 100 reactors in the United States was estimated to be very small when compared to other existing risks in society. (Levine testimony, ff. Tr. 1930, at 9).

F-35. The methodology of WASH-1400 has not been called into question since its publication; it is fundamentally sound. (Hulman et al. ff. Tr. 2091, at 6; Tr. 1974-75 (Levine)).

F-36. The Independent Risk Assessment Review Group stated in the Lewis Report (NUREG/CR-0400) that the probabilistic methodology employed in WASH-1400 was an important advance over earlier methodologies that had been applied to reactor risk and was sound. It stated that the fault-tree/event-tree approach, coupled with an adequate data base, is the best available tool with which to quantify the accident probabilities associated with nuclear reactors. (Hulman et al. testimony, ff. Tr. 2091, at 7; Levine testimony, ff. Tr. 1930, at 10; Tr. 1974-75 (Levine)).

F-37. The Lewis Report also stated that it was unable to determine whether the overall core-melt probability given in WASH-1400 was high or low and that the error bands in WASH-1400 were understated. For these reasons, the NRC stated in January, 1979 that it did not regard as reliable WASH-1400's numerical estimates of reactor accidents. (Tr. 1976 (Levine)). The Commission, in a recent communication to Congress, has stated that it accepts the Lewis Report conclusion that absolute values of the risks presented by WASH-1400 should not be used uncritically either in the regulatory process or for public purposes. In the same letter, the Commission states its support for the extended use of probabilistic risk assessment in regulatory decisionmaking. (Hulman et al. testimony, ff. Tr. 2091, at 7 and Attachment C).

F-38. The "Precursor Study" is a report, the full title of which is "Precursors to Potential Severe Core Damage Accidents: 1969-1979, A Status Report" (NUREG/CR-2497), which presents the initial results of a program performed at Oak Ridge National Laboratory and administered by the NRC. The program uses operational data in Licensee Event Reports to evaluate potential accident precursors occurring at operating reactors. The precursors are then summarized to derive a probability for severe core damage. (Hulman et al. testimony, ff. Tr. 2091, at 8; Levine testimony, ff. Tr. 1930, at 24).

F-39. The Precursor Study estimated the frequency of severe core damage accidents (averaged for all domestic light water power reactors in the decade of the 1970's) to have been between $1.7 \times 10^{-3}$ and $4.5 \times 10^{-3}$ per reactor year. In WASH-1400, the core melt frequency for the Surry plant (taken to represent pressurized water reactors) was estimated to be $5 \times 10^{-5}$ per reactor year. (Hulman _et al_. testimony, ff. Tr. 2091, at 9).

F-40. The use of WASH-1400 core melt frequency estimates is not invalidated by the Precursor Study. (Hulman _et al_. testimony, ff. Tr. 2091, at 12).

F-41. Approximately 82% of the Precursor Study's estimate of severe accident frequency comes from three events not explicitly addressed in WASH-1400, the Three Mile Island ("TMI") accident, the Browns Ferry fire and the Ranch Seco power supply failure. (Hulman _et al_. testimony, ff. Tr. 2091, at 9).

F-42. The omission of these three sequences in WASH-1400 does not invalidate the severe core damage frequency estimates of WASH-1400 as they are used in the Byron FES (with appropriate rebaselining). (Hulman _et al_. testimony, ff. Tr. 2091, at 10-12; _see_ Finding F-22).

F-43. Of these three accident or precursor sequences, only the TMI event resulted in severe core damage. While WASH-1400 did treat most elements of the TMI accident, it did not treat the possibility that the reactor operators might misdiagnose an accident in progress and turn off the safety systems necessary to cool the core. (Hulman _et al_. testimony, ff. Tr. 2091, at 10).

F-44. Operator errors of this type are less likely today than they were before the TMI accident because of regulatory requirements which have been implemented to reduce the likelihood that operators might fail to diagnose inadequate core cooling. (Hulman et al. testimony, ff. Tr. 2091, at 10).

F-45. The specific accident initiator that occurred at TMI (a transient induced loss of coolant accident) is less likely at a Westinghouse plant like Byron because the pressurizer power operated relief valves are not likely to open during feedwater transients. (Hulman et al. testimony, ff. Tr. 2091, at 10-11).

F-46. With respect to the omission of the Browns Ferry fire accident sequence, fires were not included among the accident initiators in WASH-1400. Since the Browns Ferry fire, new fire protection requirements have been developed in a new rule, 10 CFR Part 50, Appendix R. Byron Station is being reviewed against the requirements of this new rule. (Hulman et al. testimony, ff. Tr. 2091, at 10, 11).

F-47. The Rancho Seco event was caused by a power supply fault. A comprehensive analysis of the fault effects and systems interactions originating in power supplies for control and instrumentation was not done in WASH-1400. Byron and the plants studied in WASH-1400 do not appear to be as vulnerable to such faults as Rancho Seco. Byron will have additional features, such as safety-related actuation for the emergency feedwater system and safety-related auxiliary feedwater flow indication and steam generator level indication in the control room, not present at Rancho Seco. (Hulman et al. testimony, ff. Tr. 2091, at 11).

F-48. Loss of feedwater events were the fourth dominant contributor to severe core damage in the Precursor Study. Additional requirements have been implemented on all reactor plants since the TMI accident to improve auxiliary feedwater reliability. In addition, there is some likelihood that "feed and bleed" core cooling (in which decay heat removal is accomplished by adding coolant inventory with the high pressure injection system and removing decay heat energy through the safety or relief valves) could be used to prevent severe core damage.[25/] (Hulman et al. testimony ff. Tr. 2091, at 12).

F-49. The Precursor Study results do not cause a change in the population dose estimates made by the Staff in the FES. (Hulman et al. testimony, ff. Tr. 2091, at 22).

F-50. The Precursor Study, which estimates higher probabilities of severe core damage accidents than had previously been estimated, is flawed and its probability estimates are too high. (Levine testimony, ff. Tr. 1930, at 27; Tr. 2022-23, 2029 (Levine)).

F-51. The Precursor Study used generic numbers that were fed into generic event trees. Thus no account is taken of the particular plants to which the very infrequent precursor events apply or of the specific system failure probabilities that would be applicable to that particular plant. The generic approach used in the Precursor Study will almost certainly yield predicted failure probabilities that are too high. (Levine testimony, ff. Tr. 1930, at 25; see Tr. 2050-52, 2054-55 (Levine), Tr. 2163 (Newberry)).

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25/ But see Metropolitan Edison Co. (Three Mile Island Nuclear Station, Unit No. 1), ALAB-729, 17 NRC ____ (May 26, 1983), slip op. at 71-88 (Appeal Board is unprepared to state conclusively that feed and bleed will successfully provide core cooling at TMI-1, although there is some support for the position that feed and bleed can provide adequate core cooling). This is not inconsistent with testimony in this record that there is "some likelihood" that feed and bleed cooling could be used to prevent severe core damage.

F-52. The recently released Institute of Nuclear Power Operations ("INPO") analysis of the Precursor Study is properly directed to the specific plants where the precursor events occurred. This INPO report found that when the actual detailed plant configurations are taken into account, generally lower core damage probabilities are obtained, often by factors of 1/10 to 1/1000. (Levine testimony, ff, Tr. 1930, at 25).

F-53. The core damage probability estimates in the Precursor Study, not including the TMI-2 accident, average about 30 times higher than the INPO estimates. These differences are due principally to the simplified models and simplified assumptions used in the Precursor Study. (Id.)

F-54. Edison's expert, Mr. Levine, testified that the INPO estimates of severe core damage probabilities are technically superior to those of the Precursor Study, and are generally in agreement with earlier studies. (Levine testimony, ff. Tr. 1930, at 26).

F-55. The recent Sandia study regarding core melt accident risk does not demonstrate that a generic PRA is not valid to assess risk at a particular site, though it is important in such a situation to ensure that the generic study is close enough to being representative that the comparison is not misleading and to realize that the results will not be as precise as if a plant-specific study were done. (Tr. 1954 (Levine)).

F-56. The body of knowledge accumulated since WASH-1400 supports the general levels of risk reported in the FES and, in Mr. Levine's view, suggests that the values stated in the FES are conservative. (Levine testimony, ff. Tr. 1930, at 17).

(3) Treatment of uncertainties

F-57. The June 13, 1980 Statement of Interim Policy requires a discussion of "the extent to which events arising from causes external to the plant are considered possible contributors to the risks associated with the particular plant." (Hulman et al. testimony, ff. Tr. 2091, Attachment B; Tr. 6960 (Levine)).

F-58. The Commission has directed in its Statement of Interim Policy that the probabilistic risk assessment methodology be used in evaluating environmental risk. (Hulman et al. testimony, ff. Tr. 2039, Attachment B).

F-59. The Statement of Interim Policy also states that the environmental impact statements should identify major uncertainties in its probabilistic estimates. (Hulman et al. testimony, ff. Tr. 2091, Attachment B; Tr. 6961 (Levine)).

F-60. Uncertainty enters into probabilistic risk assessment in several ways. There is a factor of uncertainty in estimating the probability of various initiating events. (Tr. 6962 (Levine)). There is uncertainty in estimating the probability that various plant states will result, given certain initiating events. (Tr. 6964-65 (Levine)). There is uncertainty in estimating the probability that various release categories will result, given the existence of a certain plant state. (Tr. 6966 (Levine)). Uncertainty also enters in estimating the probability that various damage levels will result, given certain release categories. (Tr. 6967 (Levine)).

F-61. Uncertainties exist in the FES estimation of public risks in such areas as quantification of human error probabilities, the data base for component failure rates, the frequencies of external events or sabotage, and various site-dependent consequence calculations. (Levine testimony, ff. Tr. 1930, at 18).

F-62. Where the measure of uncertainty is quantifiable, the basic inputs are highly judgmental. (Tr. 6963, 6964, 6966, 6967 (Levine)). In Mr. Levine's words, "[t]here is a large amount of judgment involved in estimating uncertainties." (Tr. 6966 (Levine)). For example, Mr. Levine testified that the uncertainty associated with various external initiating events is quantifiable "mostly judgmentally." (Tr. 6964 (Levine)).

F-63. The FES includes a qualitative consideration of accident risks due to external (natural and man-caused) events, such as tornadoes, fires, earthquakes and sabotage. (Hulman et al. testimony, ff. Tr. 2091, at 3).

F-64. No quantitative assessment of accident risks from external events has been made specifically for Byron Station. (Hulman et al. testimony, ff. Tr. 2091, at 4).

F-65. No quantitative estimates of risk from sabotage were made in this case, and such estimates are considered beyond the state of the art. The consequences from successful acts of sabotage should not be different in kind from the severe accident releases estimated for internal events. (Hulman et al., ff. Tr. 2091, at 4).

F-66. The current Staff assessment of the state of the art of consideration of external event PRA methodology is that it is not sufficiently mature to produce reliable absolute estimates of risk. In other words, there are many uncertainties associated with absolute estimates obtained using current methodology; however, the estimates can often yield valuable insights if used in a relative sense. The Staff is undertaking the development of a program plan for improving the capability of external events PRA methodology. This plan is expected to be completed by early summer, 1983 and is expected to be implemented over the next 2 to 3 years. The plan is directly related to Commission planning guidance presented in NUREG-0885, Issue 2 "U.S. Nuclear Regulatory Commission Policy and Planning Guidance -1983." (Hulman et al. testimony, ff. Tr. 2091, at 5).

F-67. In developing criteria for the design of nuclear power plants, the Staff has developed considerable guidance for the treatment of external events within design bases in order to reduce substantially the risk from external events. (Hulman et al. testimony, ff. Tr. 2091, at 5).

F-68. The Staff concludes that external events can be contributors to risk, but that the state of the art in quantifying the likelihood of such events (and associated uncertainty) is not well-developed. (Hulman et al. testimony, ff. Tr. 2091, at 6).

F-69. The Staff believes that the sum of the risks from internal events and the risks from external events is within the uncertainty bounds for the internal risks. (Tr. 2240 (Hulman)).

F-70. The only cases for which external natural events have been assessed in detail are the Zion and Indian Point reactors. For Zion, the licensee has submitted a PRA which indicates that external events can be significant contributors to risk. For Indian Point, evaluations by the Staff also indicate significant risks due to external events. The best estimates of the additional risk from external events (other than sabotage) were shown to be as much as about a factor of 30 higher compared to the best estimate risks from internal events at Indian Point, and about 10 times the best estimate risk from internal events at Zion. (Hulman *et al*. testimony, ff. Tr. 2091, at 4).

F-71. While the Staff made no numerical assessment of accident risks from external events at Byron, it did draw upon information obtained from the Zion and Indian Point studies for estimates for Byron. The Staff's best estimate of accident risks from internal and external causes (exclusive of sabotage and based upon what has been learned at Zion and Indian Point) could be higher than what has been presented in the FES, but is unlikely to exceed the risk multipliers computed for Indian Point and Zion. Neither multiplier would result in risks at Byron outside an uncertainty factor of 100 times the risks from internal events. (Hulman *et al*. testimony, ff. Tr. 2091, at 4).

F-72. The uncertainty factor of 100 contained in the Staff's testimony is an attempt to quantify the upper bound of risk, based upon Indian Point and Zion data. (Tr. 2243 (Hulman)).

F-73. The Indian Point and Zion risk studies provided evidence for the first time that external events could be significant contributors to risk (Tr. 2242 (Hulman)) and suggested the possibility that the FES estimate of risk for Byron may be understated if it is based only on risk from internal events (Tr. 2247 (Hulman)).

F-74. The information on external risk at Indian Point and Zion was used because it was the only information that was available. (Tr. 2247, 2256, 2305 (Hulman)). The uncertainties associated with the external event risk estimates are very large. (Tr. 2113 (Hulman)). Such risk estimates, in Staff witness Hulman's view, are not reliable (Tr. 2112,

2121) in the sense that the direct application of such estimates at Byron would not be reliable (Tr. 2258).

F-75. The uncertainty factor of 100 is further based on work done within the Accident Evaluation Branch and is supported by other studies done by Staff members and by others. (Tr. 2250-55 (Hulman)).

F-76. Good quantification of the risks from external events and sabotage is generally considered beyond the state of the art. The Staff considered such events qualitatively in two ways. First, external events are considered within the Staff's design criteria in a deterministic manner. For example, seismicity was considered as part of the design basis evaluation of Byron. (Tr. 2304 (Hulman)). Second, engineering judgment is applied to what little information is available on external events beyond the design basis. (Tr. 2099-100 (Hulman)). Engineering judgment was applied in using the results from Indian Point and Zion to arrive at a best estimate of risk at Byron. (Tr. 2120 (Hulman)).

F-77. Despite the uncertainties discussed, the Staff believes that the FES provides a reasoned consideration of the operational risk at Byron. (Tr. 2308 (Hulman, Wohl, Newberry, Branagan)).

F-78. Mr. Levine agreed that the actual probabilities associated with the operation of Byron are within the bounds of the factor of 100 uncertainty attached to the Staff's FES analysis. (Tr. 6969 (Levine)). However, Mr. Levine did not agree that the range of uncertainty should be as large as a factor of 100 in the direction of increased risk. (Levine rebuttal testimony, ff. Tr. 6956, at 1). He testified that uncertainty stems partly from attempts to make risk prediction realistic and large values of uncertainty should not be applied where conservative estimates have been made, as in the FES. (Id. at 2; see Tr. 6992 (Levine)).

F-79. Mr. Levine considers that "there is no necessary relationship between the contributors to risk from internal events and those from external events." The ratios of external to internal risk from Indian Point and Zion can apply to Byron only "by coincidence." He believes it inappropriate to use the results of PRA's for other plants as the basis for establishing quantitative uncertainty factors to be applied to FES risk estimates for the Byron plant. (Levine rebuttal testimony, ff. Tr. 6956, at 2-3; Tr. 6970-71 (Levine)).

F-80. Mr. Levine also questioned whether it was necessary or useful to provide uncertainty estimates on the risk results in the FES, given the generalized and conservative approach taken there. He gave several reasons why he believed that the FES analysis is conservative enough so that uncertainties need not be included. (Levine rebuttal testimony, ff. Tr. 6956, at 3-7; Tr. 6968-69 (Levine)).

F-81. The Commission's rules do not require that quantitative estimates of uncertainty be provided. (Tr. 6992 (Levine)).

F-82. Putting aside the factor of 100 conservatism, Mr. Levine continued to believe that the FES estimation of risk for Byron represents a reasonable approach. (Tr. 6974 (Levine)).

F-83. The Staff's discussion of the contribution to risk from external events, and the associated uncertainty, is responsive to the requirement of the Statement of Interim Policy. (Hulman et al. testimony, ff. Tr. 2091, at 5). The approach followed by the NRC in its FES is useful because, even in using conservative estimates to account for uncertainties, it is able to show that the risks from potential accidents at the Byron Station are small compared to other risks to which the population in the vicinity of the plant are already exposed. (Levine testimony, ff. Tr. 1930 at 7).

b. Protection against severe accidents

F-84. The term "Class 9 accident" as used in Contention 62 is derived from a proposed rule change (now withdrawn) which set forth a system of classification of potential accidents for use in the Staff's NEPA assessments. Class 9 accidents were characterized in the proposed rule as ". . . involv[ing] sequences of postulated successive failures more severe than those postulated for the design basis for protective systems and engineered safety features. Their consequences could be severe. However, the probability of their occurrence is so small that their environmental risk is extremely low." (Hulman et al. testimony, ff. Tr. 2091, at 13).

F-85. The term "Class 9 accident" has often been considered synonymous with accidents involving severe releases of radioactive material to the environment, but such use is imprecise since the term "Class 9 accident" is much more inclusive. Class 9 accidents could have radiological consequences ranging from benign to severe. For example, core damage events not involving loss of containment integrity would have fairly limited radiological consequences. (Hulman et al. testimony, ff. Tr. 2091, at 14).

F-86. The term "Class 9 accident" does not necessarily mean one which leads to core damage, core melt or release to the environment. Any accident that exceeds the severity of design basis accidents is called either a "severe accident" or a "Class 9 accident." (Tr. 2160 (Hulman)).

F-87. The TMI accident has been referred to by the Staff as a Class 9 accident because it involved failures more severe than those postulated for the design basis of the plant. However, at no time were the radiological consequences of the accident to the public severe and risk to public health and safety was minimal. (Hulman et al. testimony, ff. Tr. 2091, at 15).

F-88. The design of Byron Station provides protection against a wide range of Class 9 events. (Hulman et al. testimony, ff. Tr. 2091, at 18; Levine testimony, ff. Tr. 1930, at 3, 18).

F-89. The Byron plant and its various safety systems are analytically tested for adequacy of performance against a series of design basis events ("DBE"). Each of these events imposes severe performance demands on the various safety systems which must function in response to such events to enable the plant design to satisfy regulatory requirements. Each of the events is analyzed using conservative assumptions regarding equipment availability and performance capability which are described in detail in the Staff's Standard Review Plan. Thus, the plant is tested not only against a set of challenges to its safety but under additional conservative assumptions regarding plant conditions before and during these challenges. This results in a design capability with multiple and redundant systems for coping with very severe performance demands, and provides substantial protection against unforeseen events involving multiple equipment failures and operator errors. (Hulman et al. testimony, ff. Tr. 2091, at 17).

F-90. Many plant structures, systems and components incorporated in the design to protect against design basis accidents have substantial capabilities for providing protection against more severe accidents as well. Such systems include, for example, the reactor protection and back-up shutdown systems, the emergency core cooling system, the containment building, containment spray and fan coolers, the auxiliary feedwater system and many other components and systems. (Levine testimony, ff. Tr. 1930, at 19-22).

F-91. The Applicant is developing Emergency Response Guidelines which will consider multiple failure events. In addition to the design basis events, analyses assuming various event sequences (including multiple failures) that could occur and fall outside of the required design envelope have been utilized in the preparation of the emergency operating procedures. This approach for the operators is a result of the lessons learned from the TMI-2 accident. Its objective is to further assure that the operator is able to respond to the complete spectrum of possible events. (Hulman _et al_. testimony, ff. Tr. 2091, at 17).

F-92. A margin for overall safe response to unforeseen events is provided by the flexibility incorporated in many systems and in the multiplicity of installed systems in a nuclear power plant. The plant is designed to tolerate nforeseen event sequences by appropriate use of installed dedicated emergency safety features and other equipment not considered in analysis of the DBE's. For example, alternative systems configurations may be employed or equipment may be manually actuated if automatic logic circuits do not trigger actuation. (Hulman et al. testimony, ff. Tr. 2091, at 17).

F-93. The source terms used in offsite radiological consequence analyses for many of the DBE's for Byron are based on the conservative assumptions that 100 percent of the core noble gas inventory and 25 percent of the core iodine inventory are available for release to the containment atmosphere. During the TMI-2 accident, for example, analyses of air samples indicated that a whole body dose of about 100 mrem and thyroid dose of about 15 mrem, both very small fractions of the 10 CFR Part 100 offsite radiological consequence guidelines, would have been received by a hypothetical individual at the site boundary. There is, therefore, a spectrum of severe core damage scenarios for which it can be inferred that adequate radiological protection has been provided, as long as containment integrity is maintained. (Hulman et al. testimony, ff. Tr. 1091, at 17-18).

F-94. Steps have been taken since the TMI accident to reduce the likelihood of Class 9 accidents. A number of bulletins and orders were issued, followed by the systematic formulation of a Task Action Plan containing extensive recommendations related to operator training and procedures, instrumentation, equipment reliability, and additional hardware. (Hulman et al. testimony, ff. Tr. 2091, at 18).

F-95. Requirements for licensee review of operating experience, operational quality assurance, verification of management and technical capability, verification of capability for safety review and operational advice, training of operators, review of facility procedures, review of plant maintenance capability, requirement for shift turnover procedures, requirements related to shift manning, requirements for an onsite safety engineering group, systematic assessment of licensee safety programs, requirements for a shift technical advisor all contribute to a reduction in the probability of systems failure and increased capability to take corrective actions to prevent accidents from becoming more severe. (Hulman et al. testimony, ff. Tr. 2091, at 18-19).

F-96. The effect of these changes is, first, to enhance the maintenance and operation of the systems involved in each step of identified event sequences, thus diminishing malfunction probabilities for the components of these systems. Secondly, they serve to upgrade significantly the ability of the operators and the operating organization to recognize and take the proper remedial action to cope with a malfunction should it occur. There is a combined effect from improvement in both these aspects on each and every step in the event sequence. Thus, the combined impact on the overall chance for successful safe termination of the initiating event is enhanced, and the likelihood of event sequences leading to core melt with concomitant containment failure resulting in 10 CFR 100 guidelines being exceeded is substantially reduced. (Hulman et al. testimony, ff. Tr. 2091, at 19).

F-97. The deterministic licensing requirements, based upon design basis event considerations, knowledge acquired from the TMI-2 accident, mitigative engineered safety features, multiple barriers against post-accident release of radioactivity, and additional measures, such as emergency operating guidelines which allow risk-reducing human intervention in reactor accident situations, provide reasonable assurance that the Byron plant can be operated with no undue risk to the public health and safety. (Hulman et al. testimony, ff. Tr. 2091, at 19-20).

c. Cumulative risks from multiple reactors

F-98. The possibility of cumulative doses to residents of the northern Illinois area from accidents at more than one nuclear power plant does not create undue risk to public health and safety. (Hulman et al. testimony, ff. Tr. 2091 at 22).

F-99. There has been no measured offsite radiological dose burden to northern Illinois residents due to accidents at the nuclear power plants in northern Illinois either of a discrete or cumulative nature. (Hulman et al., testimony, ff. Tr. 2091, at 21).

F-100. The likelihood of a severe accident occurring at any of the nuclear power plants in northern Illinois is sufficiently small that the addition of the Byron plants will not raise this likelihood to a significant level, even in the case of a hypothetical accident induced by an external event. (Hulman et al. testimony, ff. Tr. 2091, at 21).

F-101. Further, the likelihood of more than one severe accident at more than one plant with resultant cumulative significant radiological consequences to residents of a specific area is obviously much smaller. Its upper bound is the product of three terms: (1) the already low probability of a severe accident at one plant over its lifetime, 2) the similarly low probability of a severe accident at another plant, and 3) the probability that in each case the radioactive plume will travel over the specific area of concern, such as the DeKalb-Sycamore or Rockford areas. (Hulman et al. testimony, ff. Tr. 2091, at 21-22).

There is not presently available a radionuclide/sediment transport model which has been field-verified, for use in determining the effect of sediment and aquifer materials on radionuclide transport through the hydrosphere. In consequence, no proper NEPA analysis of this important subject can be made. In addition, as a result of this serious and unresolved problem the findings required by 10 C.F.R. §§ 50.57(a)(3)(i) and 50.57(a)(6) cannot be made.

G-2. As admitted for litigation, League Contention 109 stated as follows:

While portions of the analysis of hydrology impacts on the Rock River as a result of Byron's operation were considered at the construction permit stage, recent events indicate that C.E. has not complied with its commitments therein and has not, in any event, completed an adequate analysis in accordance with 10 C.F.R. §§ 50.34(b)(4), 50.57(a)(3)(i), 50.57(a)(6) and the requirements of NEPA. See also NUREG-0440 (NRC Liquid Pathways Study). These violations and deficiencies include, but are not limited to, inadequacies and deficiencies in at least the following areas:

(a) Studies and assessment of effects of radioactive contaminants in the sediment of the river bottom and their contribution to long term radioactivity in hydrology of the area.

(b) Long term effects of withdrawal of 30 million gallons of water a day from the Rock River, especially as related to rate of river flow needed to dilute radionuclides released from the plant in routine operation and design base or Class 9 accidents.

(c) Infiltration of groundwater and wells with radionuclides through the permeable sand and gravel deposits in the Rock River Valley, and effects on both use and supply of water for the area.

(d) The intended omissions of any groundwater models.

(e) Blockage of the flow of contaminants into the Rock River in case of serious accident. This is particularly serious in view of the quake potential of the site recently heightened by the Plum River applicability.

(f) Effects of hydrology of the Rock River of potential long term radioactivity at the plant site caused by failure to find off-site storage of spent fuel and other wastes and lack of firm plans for decommissioning of the plant.

(g) Effects of chemical discharges to the Rock River.

> (h) Analysis of long term effects of thermal plume discharge on biota of the Rock River, including but not limited to (i) effects on fish eggs and larval development, and (ii) development of pathogenic organisms such as those that cause amebic meningoenoephalitis.
>
> While this issue may have been considered at the construction phase, that hearing was a sham and in any event new facts since the construction phase call into serious question that decision. As a result the applicable findings required by the Act, NEPA, and the Regs, cannot be made herein.

G-3. By stipulation dated December 6, 1982, Contentions 39 and 109 were consolidated and rephrased as follows:

> Since the groundwater system underlying the Byron Nuclear Power Station site has not been characterized adequately, the consequences of radionuclide releases to the underlying aquifer cannot be predicted with confidence. In consequence, no proper NEPA analysis of this important subject can be made. In addition, as a result of this serious and unresolved problem, the findings required by 10 C.F.R. 50.57(a)(3)(i), 50.57(a)(6) and 10 C.F.R. 50.34(b)(4) cannot be adequately made.

G-4. Edison presented three witnesses on the liquid pathways issue. George C. Klopp is a General Design Engineer with Edison; his testimony addressed the likelihood of a core melt accident which would lead to significant releases of radioactive contaminants to the groundwater. (Klopp testimony, ff. Tr. 6750, at 1). Lawrence L. Holish is head of the Geotechnical Division of Sargent & Lundy, the architect-engineer for Byron station; he discussed the groundwater system underlying the Byron site and groundwater travel time estimates. (Holish testimony, ff. Tr. 6750, at 1). Gerald P. Lahti is Assistant Division Head of the Nuclear Safeguards and Licensing Division in charge of shielding and radiation safety at Sargent & Lundy; his testimony addressed the radiological impacts associated with assumed releases to the groundwater of radioactive contaminants.

G-5. Two members of the Staff gave testimony on the consolidated contention. Dr. Richard Codell is a Senior Hydraulic Engineer in the Environmental and Hydrologic Engineering Branch, Division of Engineering, Office of Nuclear Reactor Regulation. Gary Staley is a Hydraulic Engineer in the same branch. Their joint testimony addressed the liquid pathway impacts of operation of the Byron Station. (Codell and Staley testimony, ff. Tr. 6549, at 1-2).

G-6. The League presented Dr. Bernard John Wood as its expert witness on the liquid pathways consolidated contention. Dr. Wood is a professor in the Department of Geological Sciences at Northwestern University. His testimony focused on the question of the migration of radionuclides in groundwater underlying the Byron Station site. (Wood testimony, ff. Tr. 6879).

2. Regulatory Background

G-7. The site for a nuclear power plant must be investigated and evaluated to determine its suitability for the intended purpose. Generally, it must be demonstrated that such a site meets the requirements of NRC's regulations. One category of information that is required is a detailed description of seismic and geologic characteristics of the site. (Holish testimony, ff. Tr. 6075, at 3).

G-8. In the context of the groundwater system, it is important to investigate and identify the geologic or physical properties of the rock formation that would influence the transport of liquid radioactive materials resulting from postulated accidents. Specifically, geologic properties such as the competency of the bedrock foundation,[26/] the degree of jointing and variations in lithology are identified or characterized. Such an investigation was performed for the Byron site and the results are reported in the FSAR. (Holish testimony, ff. Tr. 6075, at 3-4).

G-9. Releases of radionuclides to the liquid pathways are one part of the analysis of severe accidents conducted by the Staff in its FES in accordance with the Statement of Interim Policy, 45 Fed. Reg. 40101 (June 13, 1980). (See Staff Ex. 2, at Section 5.9.4.5(5); Codell and Staley testimony, ff. Tr. 6549, at 6).

G-10. The liquid pathways are routes by which people can be exposed to radiation released by a nuclear power plant via surface and groundwater. Exposures involving surface water can come from drinking or swimming in contaminated water, direct radiation from contaminated shoreline sediments, and ingestion of contaminated seafood. Groundwater, which can serve as a source of drinking water, can also be contaminated. In addition, radionuclides released to groundwater can migrate to surface water. (Codell and Staley, ff. Tr. 6549, at 3).

3. Substantive Findings

a. Analyses by Edison and by the Staff

(1) Site characterization

G-11. The Byron site was investigated through a multiphased program that was structured in accordance with Sections 2.4 and 2.5 of NRC Staff

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26/ "Competency of the bedrock" refers to the ability of the bedrock to support the plant units under a static and dynamic situation. (Tr. 6782 (Holish)).

Regulatory Guide 1.70 which was in effect at the time of the investigation in the early 1970's. The results of the investigation were included first in the PSAR in 1973 and then in the FSAR in 1978. The detailed investigation included: (1) performing over four miles of geophysical surveys; (2) drilling, sampling, and selective water pressure testing of 154 borings varying in depth from 10 to 330 feet for the foundation bedrock; (3) field measurement of geologic features over the site area including outcrops and site linements; (4) installation and measurement of observation wells throughout the site and adjoining properties; (5) detailed measurements and mapping of exposed bedrock features found in the structure excavations; and (6) monitoring the behavior of structure foundations. (Holish testimony, ff. Tr. 6075 at 4-5).

G-12. Holish Exhibit 2 is a reliable characterization of the stratigraphic column of the Bryon site geologic formations. (Holish testimony, ff. Tr. 6075, at 6; Tr. 6720 (Holish)).

G-13. The four most significant hydrogeologic units at the Byron site are the glacial drift, the Galena-Platteville dolomites, the sandstone units of the Cambrian-Ordovician Aquifer (the St. Peter, Ironton and Galesville Sandstones), and the Mt. Simon Sandstone. However, only the glacial drift and the upper formations of the Galena-Platteville dolomites contribute to the calculation of aquifer travel time. (Holish testimony, ff. Tr. 6075, at 6).

G-14. The site area is covered with a mantle of glacial drift consisting mainly of glacial till covered by a few feet of loess (windblown silt). A study of borehole logs at the site indicates that the thickness of the drift averages 16 feet. Due to the generally low permeability and thinness of the till, it is not possible to develop groundwater wells by drilling into the drift. The drift is recharged by precipitation. No groundwater wells were found to exist within the glacial till, except where surface erosion exposed the bedrock allowing underground springs to occur. (Holish testimony, ff. Tr. 6075, at 7).

G-15. Beneath the thin mantle of drift are dolomites and limestones of the Ordovician-age Galena and Platteville Groups. Borehole logs indicate that the thickness of the Galena-Platteville dolomites at the site range from 100 to 225 feet below the 16 feet of glacial till. The dolomites are extensively fractured near the top 25 to 30 feet, with solutionally enlarged openings in places. These characteristics are not found at the depth where the dolomites become dense. (Holish testimony, ff. Tr. 6075, at 7).

G-16. The unweathered bedrock is at elevations 850 and below. (Tr. 6781 (Holish)). The water table at the Byron site is below the most weathered zone of the dolomite. (Tr. 6710 (Codell); Tr. 6937 (Wood)). In an April piezometric test, the groundwater levels at Byron Station were below elevation 800 (which is the level of the base of the reactor). (Tr. 6743, 6781 (Holish)).

G-17. In the site area, the Galena-Platteville dolomites are recharged by precipitation through the overlying glacial drift and discharge into the Rock River and its associated tributaries and into shallow domestic wells. (Holish testimony, ff. Tr. 6075, at 7-8).

G-18. Regionally, the Galena-Platteville dolomites are hydraulically continuous with the lower sandstone units of the Cambrian-Ordovician aquifer. However, in the vicinity of the Byron site, groundwater in the

Galena-Platteville dolomites is perched on the Harmony Hill Shale member of the Glenwood Formation which has low permeability. The low permeability of the Harmony Hill Shale member was demonstrated by comparing the hydrostatic head relationships measured in observation wells. Thus, the Glenwood Formation serves as a hydraulic barrier preventing any contamination of the lower aquifers. (Holish testimony, ff. Tr. 6075, at 8).

G-19. The next series of lower bedrock formations which comprise the regional/site hydrogeologic description supply almost all drinking water supplies for municipal uses. These water supply aquifers are hydraulically separated within the site region by the Glenwood Formation. (Holish testimony, ff. Tr. 6075, at 8).

G-20. The Glenwood Formation grades down into the thick sandstones of the St. Peter Sandstone. The Ordovician-age St. Peter Sandstone is permeable and has a relatively uniform lithology through the area. In the regional area, the St. Peter Sandstone is discharged primarily through wells for small municipalities, subdivisions, parks, and small industrial concerns. (Holish testimony, ff. Tr. 6075, at 8-9).

G-21. Lower in the stratigraphic column are the Ironton and Galesville Sandstones comprising a portion of the aquifer which is about 150 feet thick in the regional area. In the site area, the Ironton and Galesville Sandstones are about 105 to 115 feet thick. The sandstones are discharged primarily through wells serving various industries and municipalities. The Ironton and Galesville Sandstones are considered the best bedrock aquifer in northern Illinois because of their consistent permeability and thickness. Yields on the order of hundreds of gallons per minute may be obtained from the Ironton and Galesville Sandstones in wells less than 1,000 feet deep. (Holish testimony, ff. Tr. 6075, at 9).

G-22. Below the Ironton and Galesville Sandstones is the Eau Claire Formation, about 405 feet thick. The basal part of the Eau Claire Formation and the underlying Mt. Simon Sandstone (which is about 1,430 feet thick) form the basal Cambrian-age Mt. Simon Aquifer. Wells which terminate in the Mt. Simon Sandstone have yielded many hundreds of gallons per minute. (Holish testimony, ff. Tr. 6075, at 9-10).

G-23. Groundwater contamination that might emanate from the Byron Station would only travel through the glacial drift and upper formations of the Galena-Platteville dolomites. The Glenwood Formation would prevent the contamination from reaching the lower bedrock aquifers that serve as the main source of drinking water for the region. (Holish testimony, ff. Tr. 6075, at 10).

G-24. The Dunleith Formation, located within the Galena-Platteville dolomites, is the upper bedrock unit at the site. It provides topographic control and forms the foundation of the power block or safety-related structures for the Byron station. Since the Dunleith Formation is the upper bedrock unit, it has been slightly to moderately weathered. Solution activity has occurred along many of the joints, fractures and bedding planes, and reddish-brown clays or yellow silty sands may be found along these planes. The Dunleith Formation contains zones of thin green shale partings which are predominant in its lower portions. At the plant location, these shale partings grade in at the base mat elevation of the reactor vessel (approximately 66 feet below the surface). (Holish testimony, ff. Tr. 6075, at 10-11).

G-25. During foundation preparation activities for the Byron Plant, 154 borings, ranging in depth from 10 to 350 feet, were drilled. Rock samples were cored from the bedrock using 2-3/4 inch diameter double tube core barrels. Results of the drilling indicate that the Dunleith Formation is fractured, jointed, and thin-bedded, but there are no large openings along joints and bedding planes. The variation in the quality of bedrock at the plant location results from vertical variations in lithology and the proximity of the boring to principal joints which traverse the site. (Holish testimony, ff. Tr. 6075, at 11).

G-26. The most highly fractured bedrock occurs in the first 15 to 20 feet. That material has been excavated out at the Byron site. (Tr. 6743-44, 6745 (Holish). The amount of fracturing decreases dramatically with depth. (Tr. 6744 (Holish)). The bedrock is moderately to slightly fractured at the point where the Byron foundation actually sits. (Tr. 6746 (Holish)).

G-27. Four joint patterns are present in the site. They are: (1) a northwest trending pattern paralleling the regional structural trend; (2) a northeast pattern essentially perpendicular to the regional structure; (3) a north-south pathway transverse to the structure; and (4) an east-west pattern transverse to the structure. (Holish testimony, ff. Tr. 6075, at 11-12).

G-28. Based on analysis of aerial photography data, the joint patterns have a normal spacing of approximately 200 to 500 feet. Examination of bedrock exposures indicates that these four patterns are detectable below the surface, and that the spacing decreases with depth. Near-surface joint patterns mapped in the plant are reported in

Chapter 2.5 of FSAR. Some joints are clean with openings ranging from 1/16 to 1/4 inches. Some joints are clay filled due to in situ weathering and rock solutioning. Examination of outcrops and cores indicate that fracturing and weathering appears to decrease below the Dunleith-Guttenburg Formational contact. Specifically, rock quality measurement values of rock in the Dunleith are always low; whereas in the formations below the Dunleith rock, quality measurement values are higher except near areas of joints. Zones of solution activity, low rock core recovery, and low rock quality measurements have served as channelways for the movement of groundwater, and examination of the rock cores suggest that solution activity has occurred along these channelways. (Holish testimony, ff. Tr. 6075, at 12-13).

G-29. The Applicant has done extensive boring and mapping in the plant area. The field data indicate that there are extensive weathering, fracturing, jointing and bedding planes, some with sigificant solution enlargement. However, the solution enlargement decreases with depth. The significant enlargements (fractures and joints greater than 1 inch) occur mostly above the normal groundwater level and are thus not relevant to a core melt release through the basemat. Solution enlargement along bedding planes also is not a likely core melt release pathway since the extensive pressure available above the 810 level (which is about 60 feet below ground level) would cause bedding plane collapse if continuous areal solutioning were to occur. Thus only small noncontinuous solution pockets are likely along bedding planes. (Codell and Staley testimony, ff. Tr. 6549, at 15-16).

G-30. The Applicant's extensive borings at the Byron site do not show continuous large fractures or joints that would provide a direct pathway to the Rock River. (Codell and Staley, ff. Tr. 6549, at 14).

G-31. Applicant's site investigations are adequate to characterize the hydrogeologic properties of the site. (Tr. 6559 (Staley)).

G-32. Because of the dolomite bedrock characteristics, Edison decided to fill and seal all major solution enlarged joints, bedding planes, and other planar features of the bedrock by pressure rock cement grouting. The geologic descriptions indicated that grouting of the main plant area down to the Platteville-Ancell contact would significantly retard the downward and horizontal percolation of groundwater, and hence also limit the rate of solution activity. (Holish testimony, ff. Tr. 6075, at 13). The grouting extends down to the aquitard, which is the Harmony Hill shale. (Tr. 6669-70 (Codell)).

G-33. The grouting program consisted of two phases: (1) perimeter or curtain grouting; and (2) consolidation grouting of the Galena-Platteville bedrock formation to a depth of 225 feet. The objective of the curtain grouting was to establish a horizontal impermeable barrier around the entire perimeter of the Byron Station building foundation by drilling and grouting holes on spacing of 2.5 feet measured center to center. Upon completion of the curtain grouting, the second phase of consolidation grouting was started by drilling holes within the building foundation area on an initial grid spacing of 20 feet reducing to a 5 feet grid. In excess of 200,000 cubic feet of concrete grout was used. (Holish testimony, ff. Tr. 6075, at 13-14; see Codell and Staley testimony, ff. Tr. 6549, at 14-15).

G-34. The grouting of the power block area was for the purpose of eliminating any cavities or voids that might affect the plant structure under dynamic loading conditions. The environmental benefit was secondary. (Tr. 6826-27 (Holish)).

G-35. During the performance of the foundation grouting, a detailed grout injection surveillance record was maintained by experienced grouting engineers as a part of the Quality Assurance records. These records, which indicated the areas of major grout consumption, served as the basis for locating grout verification or acceptance borings. The verification borings were pressure tested and the records were compared to previous values of water pressure testing made during the initial site exploration. In situ testing of the grouted foundation rock mass verified that the grouting program resulted in making the rock mass significantly more impermeable than it had been prior to grouting. This would greatly restrict the seepage of any accidentally released effluents into the surrounding groundwater environment. (Holish testimony, ff. Tr. 6075, at 14; Tr. 6761-62 (Holish)).

G-36. The average permeabilities obtained from the pressure tests on the 19 verification holes ranged from $2.33 \times 10^{-5}$ cm/sec to $1.65 \times 10^{-4}$ cm/sec for the individual borings, and $7.45 \times 10^{-5}$ cm/sec for all 19 verification borings. In comparison, permeabilities of $2.96 \times 10^{-4}$ cm/sec and $1.05 \times 10^{-3}$ cm/sec were derived by the applicant from pumping tests made on two wells west of the plant.[27/] Thus, the grouted media beneath the plant has a permeability about a factor of 10 less than the field permeabilities. (Codell and Staley, ff. Tr. 6549, at 14-15; Tr. 6670-71 (Codell); Tr. 6828-29 (Holish)).

G-37. The grouting would increase travel time of radionuclides after a core melt to the nearest well or nearest spring "on the order of several months" or more. (Tr. 6830-32 (Holish); see Tr. 6633-34 (Staley)).

(2) Analysis of consequences of liquid pathway releases

G-38. The Staff has analyzed the consequences of a liquid pathway release at Byron in the FES. (Staff Ex. 2, Section 5.9.4.5 (5); see Codell and Staley testimony, ff. Tr. 6549, at 6).

G-39. An analysis of the potential consequences of a liquid pathway release of radioactivity for generic sites was presented in the "Liquid Pathway Generic Study" (NUREG-0440) (hereinafter "LPGS"). The LPGS compares the risk of accidents involving the liquid pathway (drinking water, aquatic food, swimming, and shoreline usage) for four conventional, generic land-based nuclear plants, and a floating nuclear plant (for which the nuclear reactors would be mounted on a barge and moored in a water body). Parameters for the land-based sites were chosen to represent averages for a wide range of real sites and were thus "typical," but represented no real site in particular. The study concluded that the individual and population doses for the liquid pathway through groundwater contamination range from small fractions to very small fractions of

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27/ See Findings G-53, G-54, G-56 (pump test data later determined not to be reliable for estimating permeability; subsequent pressure test borings yield lower permeability values).

those that can arise from the atmospheric pathways. (Staff Ex. 2, at 5-56).[28/]

G-40. There are major differences in the risk from the "liquid pathways" and (traditional) airborne exposure pathways. Probably the most significant difference between them is that much of the risk immediately following the airborne releases might be difficult to avoid (e.g., inhalation), while the risk from the liquid pathway could be virtually eliminated by avoiding contaminated water, seafood, and other activities, such as swimming or shoreline recreation. The immediate consequences from airborne exposure pathways would be difficult to avoid except by prompt evacuation of the affected population because radioactive gases and particulates would be carried at the speed of the wind, and could reach people in a matter of minutes to hours after release. There would generally be much longer delays associated with the liquid pathway, which would allow time for the monitoring and avoidance of the contaminated water. (Codell and Staley, ff. Tr. 6549, at 4-5).

G-41. It is not likely that waterborne radionuclides would pose a risk in terms of early fatalities or even early illness because the concentrations would be below the threshold levels necessary to cause immediate health effects and could be interdicted at any level deemed necessary (for example at a predetermined activity level, water supply wells in the affected aquifer could be sealed and an alternative water supply found). (Codell and Staley, ff. Tr. 6549, at 6).

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28/ Staff witness Codell took part in the LPGS. (Tr. 6551 (Codell)).

G-42. The FES discussion of liquid pathway releases is a summary of an analysis to determine whether or not the liquid pathway consequences of a postulated accident at the Byron site, initiated by a release to groundwater beneath a reactor, would be unique when compared to the generic small river land-based site considered in the LPGS. The purpose of the Staff's liquid pathway analysis is to confirm to the Staff's satisfaction that there is no unique hazard associated with the particular facility. Substantial effort went into the LPGS, which concluded that the liquid pathway relative contribution to risk was not significant. The analysis, then, for Byron uses the same types of assumptions and approximations to get a first order appraisal of the risk relative to the LPGS case. Where the risk is not significantly greater than the LPGS case, there is no need to analyze more deeply. (Staff Ex. 2, at 5-56; Tr. 6665-66 (Codell)).

G-43. The comparison is made on the basis of population doses from contaminated water and contaminated fish and direct shore line exposure. The parameters which were evaluated include the amounts and rate of release of radioactive materials to the ground, ground water travel time, sorption on geological media, surface water transport, drinking water usage, aquatic food consumption and usage of the shorelines of the involved rivers. Parameters were estimated from site-specific data wherever possible. Shoreline usage was taken as being equal per unit length of shoreline to that used in the LPGS. (Staff Ex. 2, at 5-56 to 5-57; Codell and Staley testimony, ff. Tr. 6549, at 6-7; Tr. 6557-58 (Codell)).

G-44. All of the reactors considered in the LPGS were Westinghouse PWRs with ice condenser containments. There are likely to be different mechanisms and probabilities of releases of radioactivity for the Byron reactor. It is unlikely, however, that the liquid release for a Byron reactor would be any larger than that conservatively estimated for similarly sized reactors in the LPGS. The source term used for Byron in this comparison therefore is assumed to be equal to that used in the LPGS. (Staff Ex. 2, at 5-57).

G-45. Core melt accidents are the principal contributors to risk from a liquid pathway release. (Staff Ex. 2, at 5-56).

G-46. Contaminants released in a postulated core melt accident would be initially deposited to the dolomite and limestone under the site. Groundwater flow would be initially to the southeast, east, or northeast along the gradient of the water table. This flow would take contaminants initially away from the Rock River, but it is likely that the groundwater contamination would be intercepted by small streams such as Black Walnut Creek, Spring Creek, and several unnamed drainage features and be routed to the Rock River. The nearest identified spring is approximately 1100 meters (3600 feet) from the reactors. For the purpose of the FES, it was conservatively assumed that all contaminated groundwater flowing under the site would drain into this spring and be routed to the Rock River. (Staff Ex. 2, at 5-57).

G-47. The combined drinking water, fish ingestion and shoreline exposure dose for Byron would be about a factor of three higher than those for the LPGS small river site. The Staff considers this a conservative estimate, for several reasons discussed in the FES and Staff testimony.

(Staff Ex. 2, at 5-58 to 5-59; see Codell and Staley testimony, ff. Tr. 6549, at 7, 8-10).

G-48. One of the conclusions of the LPGS was that the risks from liquid pathway releases were very small fractions of the risks from airborne release categories, even if no interdiction of the groundwater pathway was considered. Interdiction is, of course, very likely in the event of a contamination of groundwater or surface water. (Codell and Staley testimony, ff. Tr. 6549, at 8). At least two possible methods exist to reduce the flow of groundwater in the event of a radioactive release. The first method consists of pumping and retrieving contaminated groundwater ("well point dewatering") and storing it for treatment. This can be done by installing multiple groundwater wells at the perimeter of the Station's property that extend down to the Galena-Platteville formation. These wells would be close enough together so that the areas of the wells' influence overlap to create an extensive drawdown of the groundwater and reverse the hydraulic gradient. Groundwater monitoring wells also would be installed down gradient from the spill at varying elevations. The second method of retarding the flow of contaminated groundwater would involve constructing an impermeable barrier in rock. This would be accomplished through pressure rock cement grouting the entire Galena-Platteville rock formation downgradient from the spill. (Holish testimony, ff. Tr. 6075, at 26-27; Lahti testimony, ff, Tr. 6075, at 7-8; Staff Ex. 2, at 5-59).

G-49. The Staff conclusion in the FES (Section 5.9.4.5(5)) is: "The Byron Site liquid pathway contribution to population dose, therefore, has been demonstrated to be of the same order of magnitude as that predicted for the LPGS small river site. Thus the Byron site is not unique in its liquid pathway contribution to risk." (Staff Ex. 2, at 5-59; Codell and Staley testimony, ff. Tr. 6549, at 4, 6-7).

G-50. If the simplified liquid pathway site analysis had determined that the Byron site posed particularly severe consequences, a more rigorous analysis could have been performed. This did not prove to be necessary for the Byron site. (Codell and Staley testimony, ff. Tr. 6549, at 8).

G-51. Groundwater velocity and travel time was computed in the FES with the Darcy equation: $v = ki/n_e$ (where k = hydraulic conductivity, i = groundwater gradient, $n_e$ = effective porosity and v = groundwater velocity). Travel time then equals length divided by velocity. (Codell and Staley testimony, ff. Tr. 6549, at 11).

G-52. The 24.4 year groundwater travel time estimated in the FES (at 5-57) was based on permeability values from pumping test data submitted by the Applicant. (Codell and Staley testimony, ff. Tr. 6549, at 10-11).

G-53. Originally, data from field pumping tests was used to determine the water movement characteristics of the Galena-Platteville dolomite aquifer. Later, Edison reviewed this data and determined that it is not suitable for determining water movement travel time. (Holish testimony, ff. Tr. 6075, at 15).

G-54. The permeability of the aquifer has been established by comprehensive incremental <u>in situ</u> measurements of pressure testing throughout the aquifer thickness. The specific borings tested included P2 through P7, P9, P10, P15A, P22, D23 and several G series borings representative of plant site bedrock. Interpretation of the individual pressure test data and accumulation of the values from representative geologic formations yield an average permeability of 0.52 feet per day.[29/] (Holish testimony, ff. Tr. 6075, at 17). Intervenor's expert, Dr. Wood, had no criticism of the pressure test data used by Edison in calculating groundwater travel time. (Tr. 6915 (Wood)).

G-55. The effective porosity of the Galena-Platteville aquifer was determined from geophysical logging techniques used during the site exploration and compared to published values from the Illinois Geologic Survey. Effective porosity was found to vary between 2 and 10 percent. (Holish testimony, ff. Tr. 6075, at 18).

G-56. The water pressure test data submitted by Edison in lieu of the pumping test data yield lower permeability values than were used to calculate the 24 year travel time stated in the FES for the core melt scenario and would result in much longer travel times. Thus, the FES conclusion regarding the core melt release liquid pathway has not changed and other discussions in the FES which rely in part on the liquid pathway conclusion have not changed. (Codell and Staley testimony, ff. Tr. 6549, at 11-12).

G-57. Independent calculations by the Staff based on the response of the observed water table at the Byron site to the regionally-observed rate of rainfall infiltration also yield a travel time in the same range as the Staff's 24.4 year estimate. (Codell and Staley testimony, ff. Tr. 6549, at 12).

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29/ For comparison with the metric figures in Finding G-36, 0.52 feet per day corresponds to $1.83 \times 10^{-4}$ cm per second.

G-58. The Staff's witnesses testified that even if the radionuclide travel time was half that estimated in the FES, their conclusion would be unchanged since the uninterdicted population doses would still be less than an order of magnitude greater than the doses for the LPGS small river site. (Codell and Staley testimony, ff. Tr. 6549, at 13; see Klopp testimony, ff. Tr. 6075, at 12-13).

G-59. The characterization of the groundwater system underlying the Byron site has resulted in conservative calculations for groundwater flow velocities, travel times and the location of potential contamination pathways between the site and the nearest water user. (Holish testimony, ff. Tr. 6075, at 21).

G-60. Edison witness Klopp agreed with the FES conclusion that the risk from severe accidents at Byron is small and that the contribution to that small residual risk from liquid pathways is very small. (Klopp testimony, ff. Tr. 6075, at 12).

G-61. Although core melt accidents are the principal contributors to risk from a liquid pathway release, both Edison and the Staff also addressed a release to the liquid pathway from a postulated tank rupture accident. (See Lahti testimony, ff. Tr. 6075, at 3-4; Codell and Staley testimony, ff. Tr. 6549, at 14).

G-62. Edison witness Lahti addressed the possible consequences of accidental release of radionuclides to the groundwater near Byron Station as a result of a rupture of one of the 125,000 gallon boron recycle holdup tanks and as a result of a core melt accident. (Lahti testimony, ff. Tr. 6075, at 3-4). The contamination in the boron solution is fission products which are introduced into the primary coolant water by contact with the core during normal plant operation. (Tr. 6835-36 (Lahti)).

G-63. The calculated travel time in the case of the tank rupture from the point of release to the nearest well is estimated to be 30-49 years. (Lahti testimony, ff. Tr. 6075, at 5; Holish testimony, ff. Tr. 6075, at 18). The consequences of this postulated accident would be well within the 10 CFR Part 20 limits for concentrations of radionuclides in water in unrestricted areas and would not represent a threat to public health and safety. (Lahti testimony, ff. Tr. 6075, at 6).

G-64. The Staff believes that the postulated tank failure accident will not result in groundwater contamination, since the net positive differential hydrostatic head outside the building would cause flow into the buildings. (Codell and Staley, ff. Tr. 6549, at 14).

G-65. Edison witness Holish agreed that where a release occurs from a building at a point below the groundwater level, flow would be into the building rather than out. (See Tr. 6834 (Holish)).

G-66. Edison's experts also analyzed a radionuclide release postulated from a reactor core meltdown scenario using the more recent data on permeability. The determination of travel time along the postulated pathway of the effluents from the Unit One reactor core location to the nearest well is estimated by Edison to be approximately 92.7 years. (Holish testimony ff. Tr. 6075, at 25).

G-67. According to Edison's expert, the probability of a core melt accident which results in basement penetration is very small. (Klopp testimony, ff. Tr. 6750, at 6-8).

G-68. The opinion that the likelihood of basement penetration is small was based on work done on concrete analysis and on the Byron Risk Study which shows that only in a limited number of cases will basemat penetration occur. (Tr. 6779-800 (Klopp)).

G-69. According to Edison witness Klopp, the frequency of basement penetration is approximately $10^{-7}$ per reactor year for internal initiating events. (Tr. 6801 (Klopp)). Only about 1% of core melt sequences result in basemat penetration. (Tr. 6801 (Klopp)).

G-70. The shortest time estimate for basemat penetration is approximately 8 hours. This estimate assumes that the basemat is penetrated at its thinnest point (3.5 feet) beneath the reactor cavity sump. (Tr. 6803, 6798 (Klopp)). Mr. Klopp expected, however, that the vast bulk of the core material would remain in the area beneath the reactor vessel rather than migrate off to the sump some 32 feet away. At the point beneath the reactor vessel, the basemat is 11-1/2 feet thick. (Tr. 6802, 6798-99 (Klopp)). Basemat penetration, if it occurs, could take anywhere from eight hours up to three to five days. (Tr. 6764 (Klopp)).

G-71. Because of the design of Byron, which has the reactor cavity as the low point of containment, water will gather in the area in which molten core debris will accumulate, contributing to cooling. (Klopp testimony, ff. Tr. 6750, at 7-8).

G-72. Concrete ablation by the hot core occurs at a relatively slow rate. Time is available to procure an alternate source of water or restore existing systems. (Klopp testimony, ff. Tr. 6075, at 8).

G-73. There are two principal mechanisms by which liquid pathway releases could result from a core melt accident. A basemat penetration could release molten core debris to the rock under the containment. After sufficient time for cooling (estimated to be about one year), groundwater could come in contact with the debris and leach a portion of the contained radionuclides, transporting them away from the reactor. Another scenario would have the basemat penetration releasing a quantity of highly contaminated "sump" water directly to the groundwater under the containment. The latter scenario could potentially be more severe than the former, but is also much less likely. (Both of these cases were considered in the Liquid Pathways Generic Study and, by inference, in the Byron FES discussion of liquid pathway releases.) (Codell and Staley testimony, ff. Tr. 6549, at 5; Tr. 6610, 6668 (Codell)).

G-74. Edison witness Klopp testified that "more recent analysis" than the FES shows that "for the cases in which basement failure is predicted at Byron no containment water will be present." This would affect both the timing and the magnitude of radionuclide releases. (Klopp testimony, ff. Tr. 6075, at 9). Mr. Klopp considered the FES characterization of liquid pathways risk as "very significantly conservative" because: (1) the FES assumes that each and every postulated core melt event results in containment basemat failure with subsequent groundwater release; and (2) the FES assumes a sumpwater source term, which Mr. Klopp believes can be eliminated. (Klopp testimony, ff. Tr. 6075, at 10-11).

G-75. Mr. Klopp testified that ongoing work in a number of programs lead him to a conclusion that molten core debris will be coolable even outside the reactor vessel provided that an adequate supply of water is available to quench the debris and that heat removal mechanism such as containment sprays or containment fan coolers are operational. (Klopp testimony, ff. Tr. 6078, at 11).

G-76. In Mr. Klopp's view, "to have basemat penetration, one must postulate that the water needed to cool the debris is absent." He further believes that "with that water being absent, no sump water source term can exist." (Klopp testimony, ff. Tr. 6075, at 11; Tr. 6806-07; 6839-41 (Klopp)).).

G-77. Sump water results from operation of the emergency core cooling system and containment sprays. In particular, containment sprays would scrub the containment atmosphere of radionuclides, creating a sump water source term. (Tr. 6804-05 (Klopp)). The quantity of sump water could range from about 100,000 to about 400,000 gallons depending on the accident scenario involved. (Tr. 6805-06 (Klopp)).

G-78. A non-coolable debris bed is a core debris bed to which one cannot supply water or which, even given a water supply, cannot be cooled for one reason or another. For example, one could hypothesize a crust forming over the debris, isolating the remainder of the core from the coolant. (Tr. 6808 (Klopp)).

G-79. Mr. Klopp testified that a non-coolable debris bed preventing long term coolability of the core has been studied and suggested as a possibility. (Tr. 6809 (Klopp)). Other studies have concluded that such a postulate is incorrect. (Tr. 6809 (Klopp)). Mr. Klopp believed the probability of such a phenomenon to be statistically insignificant, but acknowledged that work done at Sandia estimated a greater probability. (Tr. 6810 (Klopp)).

G-80. Crusting may also form as a result of concrete decomposition. Mr. Klopp testified that such a crust 6 to 8 inches thick can form over the core as it sinks into the concrete, which decomposes at approximately

2290 degrees Fahrenheit. (Tr. 6811 (Klopp)). Small scale experiments have demonstrated this. However, he did not believe that such a crust could isolate the core from additional cooling water. (Tr. 6812-15 (Klopp)).

G-81. If the sump water was boiled off to steam and the containment had not been otherwise breached, basemat penetration would cause depressurization of the containment releasing some part of the containment atmosphere through the melthole. (Tr. 6816-18 (Klopp); see Staff Ex. 2, at 5-56). The steam inside containment would contain much of the radionuclide source term in the sumpwater. (Tr. 6807-08 (Klopp)). Some radionuclides would be released to the groundwater in this way. (See Staff Ex. 2, at 5-56; Tr. 6819 (Klopp)). Mr. Klopp had not analyzed this situation for Byron. (Tr. 6819 (Klopp)).

b. Intervenor's arguments for a more detailed liquid pathway analysis

G-82. Intervenor's expert Dr. Wood testified that because the bedrock underlying Byron Station is fractured dolomitic limestone, the Byron area is relatively more at risk in the event of meltdown than similar river sites. (Wood testimony, ff. Tr. 6879, at 6-7).

G-83. Dr. Wood did not believe that it was appropriate to treat the aquifer as a porous medium because it is "extremely fractured and jointed." (Wood testimony, ff. Tr. 6879, at 4-5).

G-84. Dr. Wood testified that the simple use of the Darcy equation can often underestimate velocities where flow is dominantly along fractures in a joint pattern in a fixed direction. (Tr. 6942 (Wood)). He gave, as an illustration, the example of a thin but continuous fracture, 1/40th of an inch wide, which would result in velocities up to 3000 feet per day in this area. (Wood testimony, ff. Tr. 6879, at 5-6; Tr. 6910 (Wood)).

G-85. Dr. Wood also cited the "many springs which flow radially from the site" and concluded that this suggests "that the occurrence of extensive fractures is high . . . . " (Wood testimony, ff. Tr. 6879, at 6).

G-86. Dr. Wood suggested that field measurements of the hydrologic properties of the dolomite aquifer should be taken using pump tests and tracer studies combined with the development of an appropriate model encompassing the entire site describing the various joint networks from the ground surface down approximately 200 feet to the Glenwood Formation. He further suggested the program be similar to the contaminant migration study made by the Illinois Geologic Survey for the Byron salvage yard site. (See generally Wood testimony, ff. Tr. 6879; Holish testimony, ff. Tr. 6075, at 20).

G-87. Dr. Wood concluded that "a much more detailed analysis of the effects of radionuclide releases to groundwater is needed" which would include "integration of field measurements of hydrologic properties with an appropriate model of the fissure system surrounding the site" and also "direct field measurement of contaminant pathways and velocities through the aquifer system underlying the sit.." (Wood testimony, ff. Tr. 6879, at 6,8).

G-88. Dr. Wood also argued for advance planning for interdictive measures. (Wood testimony, ff. Tr. 6879, at 7, 8).

(1) Fractures and joints

G-89. The most highly fractured bedrock at the site occurs in the first 15-20 feet and has been excavated out beneath the plant structures. The bedrock is moderately to slightly fractured at the point where the foundation actually sits. The amount of fracturing decreases with depth. (See Finding G-26).

G-90. The water table at the Byron site is below the most weathered zone of the dolomite. (See Finding G-16).

G-91. Extensive borings do not show continuous large fractures or joints that would provide a direct pathway to the Rock River. (See Finding G-30).

G-92. Grouting of the area beneath the power block will significantly retard the downward and horizontal percolation of groundwater. (See Findings G-32, G-36, G-37).

G-93. Staff witness Staley testified that there did not appear to be significant continuous fracturing or jointing below the containment basemat level that would lead directly to a surface water pathway. (Tr. 6655 (Staley)). This opinion was based on his review of the site mapping in the FSAR, on peizometric surface mapping information and on discussions with Staff members in the geological and geotechnical areas. (Tr. 6655-56, 6646-54 (Staley)).

G-94. The Staff concluded that the fractures and joints identified in the vicinity of the reactors are not solution enlarged and will not constitute a liquid pathway more critical than the pathway evaluated by the Staff in the FES. (Codell and Staley testimony, ff. Tr. 6549, at 16). The evidence indicates that the jointing is not continuous. (Tr. 6766-67 (Holish)).

G-95. It is possible that the presence of a continuous joint system to the river would decrease contaminant travel time, but this depends on several factors such as joint width, the presence of material in the joint,[30/] or the presence of obstructions. (Tr. 6599-600 (Staley)).

G-96. Staff witnesses were not aware of any data showing velocity figures on the order of 3000 feet per day for fractured limestone except where there are underground rivers or caves. Local geology near the Byron Station is not of the type. (Tr. 6700-01 (Codell)).

G-97. Calculations such as those done in Dr. Wood's testimony at pages 5-6 on travel time postulating a continuous fracture of given width can show "just about anything in terms of velocity" because there is no way to characterize friction losses or bend losses in the fracture using a parallel plate formula. No basis exists for meaningful assumptions for these parameters. (Tr. 6697-99 (Staley)).

G-98. When water changes direction while flowing through a crack or joint it loses energy and, therefore, velocity. This is known as a "bend loss." (Tr. 6704-05 (Staley)). Natural cracks without bends are extremely unlikely. (Tr 6705 (Staley)). Similarly, flow over a surface is impeded by the roughness of that surface. This is "friction loss." (Tr. 6707-08 (Staley)).

G-99. Dr. Wood agreed that the fractures are jointed and that one must take into account bend and friction resistance. (Tr. 6919 (Wood)).

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30/ The evidence showed that many fractures are filled with unconsolidated materials, such as illite-bearing clays, which act to sorb radionuclides. (Codell and Staley testimony, ff. Tr. 6549, at 9; Tr. 6630 (Codell)).

G-100. Dr. Wood did not know whether the springs he cited at the Byron site were discharging from the uppermost unit of dolomitic limestone or from high level bedding planes though he found their distribution consistent with the joint pattern. (Tr. 6926-27 (Wood)). If the springs were controlled by bedding plane solutioning, they would not be an expression of the joint path. (Tr. 6927 (Wood)).

G-101. Dr. Wood testified that decomposition of limestone into rubble in a coremelt would increase permeability. (Wood testimony, ff. Tr. 6879, at 7). However, Dr. Wood agreed that decomposition of the limestone into rubble would not greatly reduce travel time to the nearest spring since it only involves "a small part of the pathway." (Tr. 6912 (Wood)).

(2) Cyanide migration

G-102. Dr. Wood testified that any release of radionuclides in the Byron site area would be expected to contaminate the aquifer in a similar manner to the salvage yard cyanide contamination. (Wood testimony, ff. 6879, at 3).

G-103. On the basis of the cyanide movement at the salvage yard, Dr. Wood estimated velocities of "at least 8 feet per day." (Wood testimony, ff. Tr. 6879, at 5). He derived this figure simply by dividing the distance traveled by the elapsed time from dumping to observation. (Tr. 6888-89 (Wood)).

G-104. To Dr. Wood, the cyanide migration of 8 feet per day through groundwater invalidates the assumption that a porous media can be used to describe the site. (Tr. 6893 (Wood)). Such transport times suggest the existence of continuous fractures. (Tr. 6895, 6900 (Wood)).

G-105. The Staff does not consider the cyanide migration studies very useful. (Tr. 6601 (Codell)). The phenomena at the salvage yard site are not well-understood and the site itself is physically different than the Byron site. (Tr. 6602-03 (Codell)).

G-106. The Staff examined the available data on the cyanide containments to try to establish whether it was reliable information that could be used to analyze possible groundwater consequences of a core melt accident. The Staff rejected such use of the data, concluding that the information was not of sufficient quality for use in the coremelt release liquid pathway evaluation. (Tr. 6682-86 (Codell, Staley)).

G-107. Staff witness Codell studied all available reports on the contamination of surface and groundwater caused by the metal plating waste dumping operations near the plant site and met with Dr. Keros Cartwright and Dr. Robert Gilkeson of the Illinois State Geological Survey, who were directly involved in the studies. He concluded that it would not be appropriate to calculate the time for transport of radionuclides from core melt accidents at the plant site on the basis of the waste migration studies performed for the salvage yard. His reasons for this conclusion were that the history and location of waste disposal at the salvage yard is poorly known. Various pieces of evidence indicated that mechanisms other than hydrological may have been responsible for some of the transport of waste. Dr. Codell also found the evidence to suggest that wastes were disposed into, or very close to, several surface drainage features. These features are probably areas underlain by fractured rock, and serve to carry both ground and surface water away from the areas of heaviest contamination. There is strong evidence to suggest that much of the transport of the waste was actually by surface water. (Codell and Staley testimony, ff. Tr. 6549, at 17-18).

G-108. The reports on the salvage yard contamination point out that there is no clear indication as to how the cyanide was transported or how far it traveled. (Tr. 6692 (Staley)).

G-109. The cyanide was apparently dumped in a number of locations and in a number of different forms (i.e., in barrels, into liquid pools, as a spray). (Tr. 6674 (Codell)).

G-110. Because of the lack of information about the location and form of the waste deposited at the salvage yeard, there is no way to predict quantitatively the transport mechanisms for the cyanide. (Tr. 6702-03 (Staley)).

G-111. Dr. Wood agreed that the data on the cyanide dumping and transport mechanism "were pretty thin." (Tr. 6883 (Wood)). He conceded that one could infer that the surface water system was also involved in the transport of the contaminant. (Tr. 6883 (Wood)).

G-112. The document referred to by Dr. Wood in relation to the cyanide migration says that the cyanide was found in the shallow groundwater flow system. Dr. Wood took this to mean "tens of feet" deep. He agreed that the degree of fracturing at the plant site decreases with depth. (Tr. 6886-88 (Wood)).

G-113. If surfacewater were the operative or dominant transport mechanism at the salvage yard site, then Dr. Wood's inference of groundwater involvement (and his estimates of travel times) would be negated. (Tr. 6901, 6947 (Wood)).

G-114. There are several indications that the cyanide traveled over the surface. For example, animals died along the course of the stream. Sampling also indicated that cyanide had been dumped on the roadway. (Tr. 6770-71 (Holish)). Surface migration is also indicated by the fact that wells near the cyanide dump did not indicate the presence of cyanide while wells further downstream did. (Tr. 6771 (Holish)). There may be groundwater involvement as well. (Tr. 6772-73 (Holish)).

G-115. Surface transport of the cyanide into the stream beds or creeks apparently permitted rapid downward flow of the contaminants into the groundwater. (Tr. 6660 (Codell)).

G-116. The fact that cattle died of cyanide poisoning from drinking at pools near the salvage yard site suggests that the cyanide got into the pools by surface migration. (Tr. 6902 (Wood)).

G-117. If a contaminant travels both over a surface and through groundwater, one can only determine travel time through the groundwater if one knows how far the contaminant traveled on the surface. (Tr. 6867 (Holish)).

G-118. The contribution of surfacewater migration at the salvage yard cannot be ruled out. (Tr. 6939 (Wood)).

G-119. Edison witness Holish testified that the bedrock beneath the salvage yard site is more fractured near the surface than at the Byron Station site. This would cause a different travel time for liquid contaminants at the salvage yard site. (Tr. 6746-49 (Holish)).

G-120. With regard to the contaminant study conducted by the Illinois State Geologic Survey the investigation was concerned with a surface or near surface groundwater flow as a result of contamination from uncontrolled dumping of industrial wastes. The ISGS study was directed to measurement of surface flows with subsequent inflow and recharge to lower shallow dolomite aquifers. (Holish testimony, ff. Tr. 6075, at 23; Tr. 6612 (Codell)).

G-121. Edison's studies of the rock quality indicate the upper bedrock units are more susceptible to flow since solution and joint openings are more prevalent near the upper bedrock surface. Groundwater flow from the plant site during the postulated accident behaves in a different manner than the flows described in ISGS study for the following reasons: (1) the source of the effluent is stratigraphically lower (therefore the jointing aperture is smaller); and (2) near surface groundwater flow is directly controlled by topography, whereas subsurface flows are controlled by infiltration and recharge. The combined effect of near surface flows and varied bedrock conditions preclude the use of contaminant study data for analyzing the accidental release postulated at the Byron station. (Holish testimony, ff. Tr. 6075, at 23-24).

(3) Assumption of a porous medium

G-122. The fractured bedrock beneath the plant can be characterized as equivalent to a porous medium. This conclusion is based on examination of the bedrock and the location, orientation and approximated aperture size of the various joints. (Tr. 6864-66 (Holish)).

G-123. The Staff considered other models but decided that using an assumption of a porous medium is the best assumption available for the aquifer underlying the Byron site. (Tr. 6595-96 (Staley)). To model a system of fractures would be extremely difficult and timeconsuming. (Tr. 6596-97 (Staley)). Borings and pressure testing provide a good representation of the flow situation at the Byron site. (Tr. 6596 (Staley)).

(4) Tracer tests

G-124. Tracer study is the only way to determine travel time in the bedrock unequivocally. (Tr. 6919 (Wood)). If the travel time of approximately 30 years is correct, the tracer test would take the same 30 years to complete. (Tr. 6920-21 (Wood); see Tr. 6768, 6841, 6862 (Holish)).

G-125. While more data are always better, enough reliable data are present to permit a conclusion to be drawn that groundwater flow is not extremely fast. (Tr. 6644 (Codell)). This includes data on permeability, the lack of surface drainage fractures, and water table behavior. (Tr. 6644-45 (Codell).

(5) Planning for interdiction

G-126. Interdictive measures have never been implemented in the context of a nuclear power plant release because there have been no substantial releases. (Tr. 6570-71 (Codell)). Analogues, such as hazardous waste isolation, suggest that interdictive measures can be used successfully. (Tr. 6570, 6573 (Codell)).

G-127. Specific interdictive measures to be utilized would depend on the conditions of the accident. (Tr. 6635 (Codell)). For example, a decision on where to put a grout curtain would depend on time estimates for the migration of the molten core through the area already grouted and for insertion of the new grout curtain. Those decisions must be made when the particular situation occurrs. (Tr. 6637-38 (Staley)).

G-128. Grouting for foundation purposes is not unusual. (Tr. 6638 (Staley)). The methodology for grouting is standard. (Tr. 6640 (Staley)). Routine engineering work is involved in this process. (Tr. 6641 (Staley)).

G-129. Edison witness Klopp has had personal experience with interdictive measures concerning radioactive contamination. (Tr. 6727-30 (Klopp)). In his view, development ahead of time of a specific plan for interdictive measures would reduce flexibility of response and would lead to increased exposures. (Tr. 6730 (Klopp)).

(6) Liquid versus airborne pathways

G-130. Dr. Wood cited a document, NUREG/CR-1596, for the proposition that "the latent effect of a liquid pathway release in the case of a meltdown could be comparable to the effects from airborne releases." (Wood testimony, ff. Tr. 6879, at 6).

G-131. Dr. Wood himself has done no study or analysis of the consequences of severe accidents at nuclear power plants or the latent effects of liquid pathway releases in the case of a meltdown. (Tr. 6872-73 (Wood)).

G-132. NUREG/CR-1596 acknowledges by its own terms that its estimates of consequences of hydropheric releases may be somewhat misleading because it assumes that no interdictive procedures are used. (Tr. 6930 (Wood)). If interdictive measures were taken, the hydrospheric contamination would be much less than through the atmosphere. (Tr. 6930 (Wood)). Under any circumstances, some interdiction would be anticipated. (Tr. 6932 (Wood)).

G-133. Several reasons exist why one would expect the consequences of release via the atmospheric pathway to be greater than via the liquid pathway. Radionuclide dispersal would be much more rapid through the atmosphere. There is a greater likelihood of acute health effects from the atmospheric pathway release. Interdictive measures are more difficult in the case of an atmospheric pathway release. (Tr. 6932-33 (Wood)).

G-134. The assumptions underlying NUREG/CR-1596 -- instantaneous release to the groundwater of all radioactivity in the sumpwater and the melt debris and no interdiction -- are extremely conservative; their use yields a bounding analysis with no basis in reality. (Tr. 6740, 6823-24 (Klopp)).

(7) Conclusion as to Intervenor's arguments

G-135. The type of studies and modeling suggested by Dr. Wood are not warranted for the investigation of the Byron site. (Holish testimony, ff. Tr. 6075, at 20-21).

G-136. Dr. Wood could not say whether Byron Station presented more of a hazard in terms of liquid pathway releases than any other reactor site in the United States. While he cited the permeable limestone at Byron, he was not familiar with the geology of any of the other sites and could not say that Byron presents a special case. (Tr. 6933-34 (Wood)).

G-137. The Board concludes that the groundwater system underlying the Byron site has been characterized adequately, that the Byron site is not unique in its liquid pathway contribution to risk and that there would be ample time to intercept accidental groundwater releases before they could contaminate surface water or downgradient wells. (See Codell and Staley testimony, ff. Tr. 6549, at 22).

G-138. The Board further concludes that the consequences of radionuclide releases via the liquid pathway have been adequately and conservatively addressed in the Staff's FES and a more detailed liquid pathway analysis is unwarranted.

## IV. CONCLUSIONS OF LAW

Based on the entire evidentiary record of this proceeding and upon the foregoing findings of fact, the Board concludes the following:

1. Contrary to the claim in League Contention 8, a reasoned consideration of the risks associated with operation of Byron Station has been presented in the Final Environmental Statement in conformity with the requirements of NEPA and the Commission's Statement of Interim Policy.
2. Contrary to the claim in League Contention 62, the design of Byron Station does provide substantial protection against so-called Class 9 accidents and does provide reasonable assurance that the activities to be authorized by the operating license can be conducted without endangering the health and safety of the public.
3. Contrary to the claim in DAARE/SAFE Contention 2A, the potential for cumulative dose effects from discrete accident events at plants in northern Illinois, including Byron Station, does not pose an unreasonable level of risk to the health and safety of area residents.

4. Contrary to the claim in the consolidated liquid pathways contention (former League Contentions 39 and 109), the hydrogeology of the Byron Station site has been characterized adequately and the consequences of a release of radionuclides via the liquid pathways have been reasonably and adequately addressed in the FES and do not pose an undue risk to the health and safety of the public.

## V. ORDER

WHEREFORE, in accordance with the Atomic Energy Act of 1954, as amended, and the Rules of Practice of the Commission, and based on the foregoing Findings of Fact and Conclusions of Law, IT IS ORDERED THAT this Supplemental Partial Initial Decision shall constitute a portion of the ultimate Initial Decision to be issued upon resolution of the remaining contested issues in this proceeding.

IT IS FURTHER ORDERED, in accordance with 10 CFR §§ 2.760. 2.762, 2.764, 2.785, and 2.786 that this Supplemental Partial Initial Decision shall become effective and shall constitute, with respect to the matters addressed herein, the final decision of the Commission thirty days after the date of issuance hereof, subject to any review pursuant to the above cited rules of practice. Exceptions to this decision may be filed within ten (10) days after service of this supplemental partial initial decision. A brief in support of such exceptions must be filed within thirty (30) days thereafter, forty (40) days in the case of the Staff. Within thirty (30) days after service of the brief of Appellant, forty days in the case of the Staff, any other party may file a brief in support of, or in opposition to, such exceptions.

THE ATOMIC SAFETY AND LICENSING BOARD

Ivan W. Smith, Chairman

Dr. Richard F. Cole, Member

Dr. A. Dixon Callihan, Member

Dated at Bethesda, Maryland
this day of , 1983

Richard J. Rawson
Richard J. Rawson
Counsel for NRC Staff

Dated at Bethesda, Maryland
this 1st day of July, 1983

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 |
| | ) | 50-455 |
| (Byron Station, Units 1 and 2) | ) | |

CERTIFICATE OF SERVICE

I hereby certify that copies of "NRC STAFF PROPOSED FINDINGS OF FACT AND CONCLUSIONS OF LAW IN THE FORM OF A SUPPLEMENTAL PARTIAL INITIAL DECISION ON CLASS 9 ACCIDENTS AND LIQUID PATHWAY RELEASES" in the above-captioned proceeding have been served on the following by deposit in the United States mail, first class, or, as indicated by an asterisk, by deposit in the Nuclear Regulatory Commission's internal mail system, this 1st day of July, 1983:

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