Early Site Permit Meeting with Nuclear Energy Institute

March 5, 2003
NRC Handouts

NRC Escorting Requirements

Code Yellow

- Visitors MUST be escorted at all times except on the lobby level
- If you leave this room for ANY reason, you must have an NRC employee with you
- We appreciate your cooperation

Conceptual Framework for the Part 52 Process

- "In 1989, the NRC completed a long-term initiative to produce an alternative to the traditional 10 CFR Part 50 licensing process. The objectives of this initiative - - Part 52 rule - - are to resolve licensing issues early and to enhance the safety and reliability of nuclear power plants. The new regulations do not compromise safety, do not reduce the scope of issues evaluated by the NRC, do not reduce the openness to public review and challenge during the licensing process."

- Remarks by Ivan Selin
Chairman, U. S. Nuclear Regulatory Commission
Before the
Early Site Permit Demonstration Siting Conference
Arlington, Virginia
October 14, 1992

Response to NEI Comments on ESP-3

- **Staff Response to NEI Comment 1 (See Attachment)**
  - ESP finality provisions require that the staff have assurance that the ESP data will be acceptable to support design and operation of future safety-related SSCs.
  - Appendix B constitutes the NRC standard for ensuring adequate QA for safety-related SSCs.
  - QA-related findings will pertain to whether the ESP data supports a substantive regulatory finding and future use of the data for COL purposes.

Response to NEI Comments on ESP-3 (Cont'd)

- Staff Response to NEI Comment 1 (Cont'd)

  - Based upon the above logic, the staff QA review to ensure "equivalent in substance" QA controls provides flexibility to focus on substantive findings yet retains a uniform review standard consistent with the staff review of comparable safety-related activities in the current fleet of nuclear power plants.

  - The staff is developing a QA section in RS-002, ESP Review Standard, to provide review guidance for this subject. This section will be issued for public comment by the end of April 2003.

Response to NEI Comments on ESP-3 (Cont'd)

- Staff Response to NEI Comment 2 (See Attachment)
  - QA controls pertain to all of those planned and systematic actions necessary to provide adequate confidence that safety-related SSCs will perform satisfactorily in service. Therefore, to the extent that site-related information would affect the performance of future safety-related SSCs, QA controls will be applicable to that information.

Response to NEI Comments on ESP-3 (Cont'd)

- Staff Response to NEI Comment 3 (See Attachment)

  - During the June 2002 meeting with NEI, the ESP applicants agreed to transmit their QA program descriptions to the NRC staff for review.

  - The NRC RAI process is a disciplined process directed by procedure to obtain the information needed to support staff findings and conclusions. As described in Item 6 of the February 3 Staff Letter, the staff will utilize RAIs to request information related to the quality controls necessary to provide reasonable assurance of the integrity and reliability of data that will affect the performance of future safety-related SSCs. Applicants may choose to submit a description of those QA controls (e.g., QA program description) to improve the efficiency of the staff review.

Response to NEI Comments on ESP-6

- Staff Response to NEI Comment 1 (See Attachment)
  - Engineering judgement supported by regulatory guidance will be used to determine that PPE values are not unreasonable.
  - Although neither the PPE Worksheet or its companion definition document will be provided with or form a part of the ESP Applications, it is necessary for PPE values conform to or agree with fact, logic or known truth.

Response to NEI Comments on ESP-6 (Cont'd)

- Staff Response to NEI Comment 2 (See Attachment)
  - Identification of technical parameter margins is a good engineering practice.
  - It is expected that an ESP applicant will provide an explanation as to the origin of the PPE values and how those values were selected for their application.
  - The February 5 Staff Letter establishes the staff's expectation that the margin described in the December 20, 2002 NEI Letter on ESP-6 will be disclosed; otherwise the staff will assume that margin considerations are not applicable.

Response to NEI Comments on ESP-6 (Cont'd)

- Staff Response to NEI Comment 3 (See Attachment)
  - The definition applicable for correctness in Item 6 of the Staff February 5 Letter is a "conformance to a set figure (or in this case a known design)." The other definition for correctness as "conformance to or agreement with fact, logic or known truth" is applicable to staff reviews which verifies that PPE values are not unreasonable.
  - Given the role of the PPE to act as surrogate design information, these values must be sufficient for the required safety and environmental reviews. PPE values will be determined by the applicant. The staff will assess the completeness of those PPE values for safety and environmental impacts.

Response to NEI Comments to ESP-6 (Cont'd)

- Staff Response to NEI Comment 4 (See Attachment)
  - RS-002, Draft ESP Review Standard, specifies that ESP applicants address environmental impact mitigation alternatives. Public comments should be submitted by 3/31/03 for staff consideration of recommended changes.

Response to NEI 2/12/03 Approach for ESP-7

- Staff comments on Discussion Paper entitled "ESP-7 Approaches to Performing Radiological Consequence Assessment of Design Basis Accidents for Preparation of an Early Site Permit - Site Safety Assessment (See Attachment)

  - Generally consistent with existing staff position.

  - Item 1 - Replace "(95 percent)" with (Regulatory Guide 1.145).

  - Second paragraph, page 3. Delete reference to "NUREG-0800, Standard Review Plans" and substitute with Section 15.0, "Radiological Consequences of Design Basis Accidents" of RS-002, Draft ESP Review Standard (to be issued by April 2003).

Response to NEI 2/12/03 Approach (Cont'd)

- Staff comments on the summary statement entitled "Follow up to NRC Letter on ESP-7 for Discussion with NRC on March 5, 2003." (See Attachment)

  - Item 1 states that "the site (χ/Q) approved in an ESP will not be subject to further NRC review at COL."

    Without a definite plant design and its location on a site, the fission product release points are not known and the distances to Exclusion Area Boundary / Low Population Zone may change. At the COL stage, the staff will review to confirm that the proposed design bounds the site (χ/Q) parameter approved in the ESP. [The above comments are consistent with Items 8 and 9 of the February 5 Staff Letter on ESP-6]

  - Item 2 is consistent with the staff position.

Response to NEI 2/12/03 Approach (Cont'd)

- Staff comments on the summary statement (Cont'd)
  - Item 3 - During the July meeting, NEI described a set of bounding PPEs that would be established including release points and a postulated source term. Information would be obtained on the atmospheric dispersion (χ/Q) characteristics for the selected site. Multiple source term PPE values can be selected to represent the range of reactor designs considered by the applicant (See PPE Worksheet for examples).
  - Any PPE value would be a term and/or condition to the ESP and the source term PPE values would be reviewed similar to other PPE values. The focus of the staff review is to assess PPE values for safety and environmental impacts.

Response to NEI 2/12/03 Approach (Cont'd)

- Staff comments on the summary statement (Cont'd)

  - Last paragraph states in parts " times and rates of fission product appearance in containment and the isotopic quantities and the chemical form of fission product released to the environment are not necessary to support... and will not be provided."

    Times and rates of fission product appearance in containment and the chemical forms of fission products for DBAs are needed for the staff to determine the reasonableness of the PPE source term values. The isotopic quantities are needed as the dose conversion factors are based on each individual isotope as shown in PPE Worksheet, Rev. 0. The chemical forms of fission products are also needed for severe accident analyses to determine the fission product ground deposition rates.

Response to NEI Comments on ESP-11

- Staff Response to NEI Comment 1 (See Attachment)

  - ESP-11 pertained to "the need for a common understanding of the circumstances under which an ESP would be approved for a period less than twenty years."

  - Therefore, the subject sentence pertains to the narrow question in ESP-11 regarding ESP duration assessment, not the larger question as to the utility of the ESP for Combined License (COL) purposes.

Response to NEI Comments on ESP-11 (Cont'd)

- Staff Response to NEI Comment 2 (See Attachment)
  - The last sentence in Item 4 of the February 5 Staff Letter cites geology and meteorology as examples in a RAI process for the duration issue. There is no specific concern identified at this time.

ATTACHMENT

NEI Advance Briefing Materials E-mailed to the NRC from 2/21/03 - 3/4/03

Follow-up Discussion of NRC Responses to ESP Issue Resolution Letters

ESP-3

The industry has concerns with the NRC's Feb. 3 response on ESP-3, QA Requirements for ESP. In particular, staff response is requested on the following items:

1. We are concerned that the resolution of the central QA issue documented by the NRC staff in its Feb. 3 letter on ESP-3 is not as discussed in our Dec. 5 public meeting and, in fact, provides less clarity than the staff's Dec. 5 position statement. In particular, staff's Feb. 3 position that quality controls applied to ESP activities associated with site safety should be "equivalent to the controls specified in Appendix B" is at best confusing and at worst in conflict with the staff's position that an Appendix B program is not required for ESP.

   Revision of IMC-2501 presents an opportunity to provide the clarity on this matter that has been the focus of extensive industry – NRC discussions since April 2002. As it prepares to revise IMC-2501 to conform to the resolution of this issue, we urge the staff to use alternative language to describe QA requirements and expectations related to ESP. Appropriate alternative language is identified in the industry comments on IMC-2501.

2. We are concerned at the expansion of the scope of Appendix B indicated by the staff's Feb. 3 letter where it says Census and NOAA information "should be controlled using processes for maintaining data integrity, traceability, document control, evaluation, analysis, and record storage that are equivalent to the processes and controls described in Appendix B." For example, population projections based on Census data are not performed using safety-related codes, nor are they performed under Appendix B or processes equivalent to Appendix B.

3. Item 6 of the staff's Feb. 3 letter states, "If a description of the [QA] controls is not submitted with the ESP application, these evaluations will be facilitated through RAIs ...." Please clarify the staff's intent, in particular, that the staff does not intend to request QA program descriptions via RAI if those descriptions are not provided in the ESP application.

ESP-6

1. PPE values reflected in ESP applications will be based on vendor-supplied information on existing and future reactor designs. What criteria will the staff

use to determine "that the PPE values are not unreasonable?"

2. Item 4 of the staff's Feb. 3 letter suggests that differences between PPE values in ESP applications and PPE Worksheet values (due to application of margin) could cause ambiguity or confusion. Clarification is in order to recognize that the PPE Worksheet was provided generically by NEI for pre-application information only and has no relevance to NRC review of a specific ESP application, except as background information on the PPE approach and process for selecting PPE values.

   The industry considers the inclusion of margin in PPE values to be at the discretion of ESP applicants. This is consistent with the view that PPE values will not be reviewed for correctness. Please clarify the staff's expectation in Item 4 regarding identification of margin.

3. Item 6 acknowledges that the staff will not review PPE values for correctness. Please clarify Item 5 where it says, "the NRC will review to determine whether the PPE values are sufficient to enable the staff to conduct its required review." Does this staff statement pertain to the completeness of the parameters included in the PPE or to the values of parameters or both?

4. The NRC's Feb. 5 response did not address the second part of Item 11 of our Dec. 20 resolution letter on ESP-6, regarding consideration of environmental impact mitigation alternatives. As discussed in Enclosure 1 of our Dec. 20 letter, ESP applications will identify the scope of mitigation alternatives considered and include additional information beyond that contained in the PPE to support NEPA required reviews in this area. To the extent not addressed in the ESP, COL applications must include evaluations of mitigation alternatives for environmental impacts determined to be significant.

   Please provide the staff response regarding consideration of environmental impact mitigation alternatives.

ESP-11

1. ESP applications will be based on 60 years to account for a 20-year term of the ESP plus the 40 year initial licensing term of an operating reactor. Please clarify the last sentence in Item 2 of the staff's Feb. 5 letter on ESP-11.

2. Please elaborate on the last sentence in Item 4, which suggests some expectation on the part of the staff for ESP applications to include specific justification that meteorological, geological and similar analyses are valid for 60 year analysis in an ESP application (20 year permit plus 40 plant life). Is there a specific concern about meteorological and geological analyses?

ESP-7 Approach to Performing
Radiological Consequence Assessment of Design Basis Accidents
For Preparation of an Early Site Permit
Site Safety Assessment

Pursuant to your February 5, 2003, letter regarding Nuclear Energy Institute's (NEI) proposed resolution to Early Site Permit Topic No. 7 (ESP-7), "Guidance for Satisfying 10 CFR 52.17 (a) (1) Requirements," NEI has reviewed your position and has prepared an approach that will demonstrate how the ESP Applicants will meet the requirements of 10 CFR 50.34 (a) (1). This approach uses bounding reactor accident source terms and post-accident site dispersion factors (χ/Q) to evaluate the acceptability of an ESP Applicant's site. The proposed approach is similar to that outlined and discussed with the Nuclear Regulatory Commission (NRC) staff in our December 5, 2002 meeting for addressing design basis accidents for an ESP Applicant's Environmental Report.

10 CFR 52.17 requires that the ESP application include a safety assessment of the site that includes an analysis and evaluation of the major structures, systems, and components of the facility that bear significantly on the acceptability of the site under the radiological consequence factors identified in 10 CFR 50.34(a)(1). 10 CFR 50.34(a)(1) provides general requirements for demonstrating this objective. 10 CFR 52.17 requires that the site characteristics comply with 10 CFR Part 100. Part 100 identifies requirements for the site (atmospheric dispersion) characteristic in 100.21(c) related to radiological dose consequences of postulated accidents. The approach described here is sufficiently flexible to be used with the Plant Parameters Envelope approach in lieu of specific design information or when several candidate reactor plants are being considered for the ESP site. The approach to be used for site safety assessments follows.

1. **Introduction:** Applicants for Early Site Permits will evaluate a spectrum of representative design basis accidents in order to assess the radiological consequences associated with the alternative advanced reactor technologies being considered for future deployment. The selection of accidents will be based upon current NRC regulatory guidance to the extent practical. Short term accident (95 percentile) site dispersion factors at the exclusion and low population zone boundaries that are based on onsite data will be used to perform the assessments. The design basis accident(s) radioactivity released to the environs will be as provided in the reactor vendor's standard safety analysis reports or as specified to be bounding by the reactor supplier. These released activities are indicative of the performance of plant's major structures, systems, and components intended to mitigate the consequences of accidents.

2. **Selection of Accidents:** Accidents will be selected to cover a spectrum of events and reactor types. Consistent with regulatory objectives for determining site suitability, the selection will include low probability accidents postulated to result in significant releases of radioactivity to the environs. As such it is expected that the evaluations will include light water reactor (LWR) Loss of Coolant Accidents (LOCA) that presume substantial fuel damage in the core followed by the release of significant amounts of fission products into a containment building. In addition, accidents of higher frequency but with lower potential for significant releases will be considered as part of the site evaluation to permit qualitative assessment of potential risks at the site. Accidents identified in Regulatory Guide 1.183, vendor design certification packages, vendor technical summary documents, and USNRC standard review plans for safety analyses are expected to be included as part of the evaluation of LWRs.

February 21, 2003

Follow-up to NRC letter on ESP-7 For Discussion w/NRC on March 5, 2003

The attached discussion paper reflects industry consideration of the NRC's Feb. 5 letter on ESP-7. This revised ESP-7 approach generally reflects the approach suggested by Item 7 of the staff's ESP-7 letter. Before sending another ESP-7 resolution letter, we would like staff feedback at our March 5 public meeting. In particular, we would like NRC staff feedback on the following understandings and expectations, which we would seek to establish in a second ESP-7 resolution letter (superceding our letter of Dec, 20, 2002) that would be sent to the NRC after the March 5 meeting:

1. As part of ESP applications, the NRC staff will review and approve the site's short term atmospheric dispersion factors (χ/Qs) to be used in future COL applications. The site (χ/Qs) approved in an ESP will not be subject to further NRC review at COL.

2. ESP applications will provide bounding DBA dose consequence analyses to demonstrate that the site is acceptable based on the radiation dose consequence factors of 10 CFR 50.34(a)(1).

3. NRC review and approval of source term information is not part of ESP. Similar to PPE values, bounding release history provided in ESP applications will not be reviewed for correctness.

4. If a COL application references an ESP and a certified design, the COL applicant must provide information sufficient to demonstrate that the design χ/Q falls within (i.e. is greater than or equal to) the χ/Q specified in the ESP.

5. If a COL application does not reference a certified design, the COL applicant must perform an analysis and evaluation using the site χ/Q demonstrating acceptability under the radiological consequence evaluation factors of 10 CFR 50.34(a)(1).

The approach described in the attachment generally reflects the approach suggested by Item 7 of the staff's Feb. 5 letter on ESP-7. However, we note that times and rates of fission product appearance in containment and the isotopic quantities and the chemical forms of fission product released to the environment are not necessary to support bounding ESP dose consequence analyses and will not be provided. Bounding dose consequence analyses to be provided in ESP applications are indicative of the major SSCs intended to mitigate the consequences of accidents and thus satisfy the 10 CFR 52.17(a)(1) requirement to describe the major SSCs of the facility that bear significantly on the acceptability of the site under the radiological consequence evaluation factors in 10 CFR 50.34(a)(1).

Discussion paper attached.

It is not considered necessary to analyze all possible accidents for the alternative reactor types that may be deployed. The set of accidents to be reviewed is expected to focus more on the light water reactor (LWR) designs because they have pre-approved, certified standard designs and have recognized postulated accident bases. Accidents of lesser severity for some of the newer reactor types being considered are not as well defined and application of the accepted analytical conservatisms applied to LWRs through regulatory guides and standard review plans may be inappropriate based on their design characteristics. In addition, the newer reactor designs are being developed because of their potential for inherent safety and reduced radiological consequences relative to current LWR technology (for example, multi-module station installations for the GT-MHR and PBMR, and elimination of the LOCA in the IRIS by eliminating all large loop piping). Thus, it is not necessary to consider all accidents but only a sufficiently robust and conservative set in order to demonstrate site suitability.

2. **Source Terms:** Time-dependent activities released to the environs are used in the dose evaluations. The released activities account for the reactor core source term and accident mitigation features in the reactor vendor's certified designs. For reactor designs not currently certified, conservative releases specified by the vendor will be used in the evaluations.

   The different reactor technologies have used or planned to use different source terms and approaches in defining the accident spectrum and the associated activity releases. For example, the ABWR certified design source term is based on the use of TID-14844 methodology whereas the AP1000 makes use of the alternate source term guidance and NUREG 1465 methodology (as discussed in Regulatory Guide 1.183). The ACR-700 uses attributes of both methods in assessing their limiting design basis scenarios. The AP600 and or AP1000 source terms and releases are expected to bound the limiting accident release for the IRIS advanced reactor. The GT-MHR and PBMR use mechanistic accident source terms considered representative of gas-cooled reactors and are bounded by the LWR technology. Because of the different source terms and release mechanisms, the ESP accident assessment will identify the source terms and bounding radioactivity releases for the various reactor technologies and spectrum of design basis accidents considered.

3. **Evaluation of Radiological Consequences:** Accident doses will be evaluated at the site's proposed exclusion area boundary and low population zone. The evaluations will use short-term accident dispersion characteristics based on Regulatory Guide 1.145 methods and on-site meteorology data. Accident doses will be expressed as total effective dose equivalents (TEDEs) consistent with the requirements of 10 CFR 50.34. The doses will be determined using accepted dose conversion factors, breathing rates, and time intervals such as those incorporated into regulatory guides and the Standard Review Plans.

   The site safety analysis will identify to the extent practical the activity release paths, credited mitigation features, significant analysis parameters and assumptions, and the time-dependent activities released to the environment. Where dose consequences have been determined for certified designs using approved methods and representative site meteorology, the consequences will be scaled to the proposed site using the site-specific χ/Q dispersion characteristics.

The doses will be compared to the acceptance criteria in 10 CFR 50.34 and 10 CFR 100 to demonstrate that an adequate level of protection to the public can be maintained. In this context, the offsite radiological consequences would be considered acceptable if the doses are below the 10 CFR 50.34 criteria as defined in Regulatory Guide 1.183 and the NUREG-0800 Standard Review Plans. In the absence of specific dose limits for non-LWR accidents, the LWR dose criteria would be applied based on consideration of frequency of occurrence and the consequences of the non-LWR accident.

The ability to meet the criteria of 10 CFR 50.34(a)(1) with margin provides assurance that an acceptably low risk of public exposure exists at the ESP site. The integration of surrogate design information via the PPE concept and site characteristics in the assessment of accident radiological consequences in this approach is considered sufficient to demonstrate compliance with the 10 CFR 52.17(a)(1) requirements to evaluate the major structures, systems and components of candidate reactors that bear significantly on the acceptability of the site.

Dr. Ronald L. Simard
SENIOR DIRECTOR, BUSINESS
SERVICES DEPARTMENT
BUSINESS OPERATIONS DIVISION

DRAFT C – 2/24/03

Mr. James E. Lyons
Director, New Reactor Licensing Project Office
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: Resolution of Generic Topic ESP-8 (Tables S3 and S4)

Project 689

Dear Mr. Lyons:

In several public meetings between September 25, 2002 and January 29, 2003, we discussed generic early site permit topic ESP-8, which concerns the methodology that ESP applications will use to determine the environmental impacts associated with the fuel cycle and transportation activities for Early Site Permit application environmental reports. This methodology uses the following general approach:

- As in the WASH reports that form the basis for the current Tables S3 and S4, use conservative but reasonable assumptions
- Compare ESP fuel cycle and transportation requirements with those assumed to calculate environmental impacts shown in Tables S-3 and S-4
- Evaluate any potential increases in fuel cycle requirements [e.g., enrichment]
- Demonstrate that the current values shown in Tables S3 and S4 are suitable for determining the expected fuel cycle environmental impacts in ESP applications or provide new impact values if appropriate

Attachment 1 to this letter provides examples and shows how this methodology would be applied. Where the ESP fuel cycle requirement is less than the value assumed for Tables S3 or S4, the table values are bounding for ESP environmental impact evaluations. Where an ESP fuel cycle requirement is higher than the value assumed to prepare Tables S3 or S4, additional evaluations will be performed to determine whether advances in fuel cycle technology can be shown to limit the associated environmental impacts to less than the Table S3 and S4 values. Additional evaluations would only be performed if technology advances cannot be shown to limit environmental impacts to the Table S3 and S4 values.

Our ESP-8 discussion focused primarily on the methodology we plan to use to determine whether Tables S3 and S4 are appropriate for estimating the environmental impacts for ESP applications for the fuel cycle and transportation activities. We request that, by reply to this letter, the NRC confirm the understandings and expectations identified below.

1. Subject to NRC review in ESP applications, mitigating factors associated with modern fuel cycle and transportation practices may be credited in evaluations to demonstrate that environmental

impacts identified in Tables S3 and S4 are representative of fuel cycle and transportation impacts for several types of advanced reactors being evaluated in connection with pilot ESP applications.

2. The methodology described in the enclosure to this letter is appropriate for evaluating additional reactor technologies in a future COL application.
3. A COL applicant proposing to build one of the reactor types evaluated against Tables S3 and S4 in a referenced ESP must confirm that fuel cycle and transportation requirements fall within those evaluated for ESP.
4. A COL applicant proposing to build other than one of the reactor types evaluated in a referenced ESP may use this methodology to demonstrate that fuel cycle and transportation environmental impacts for the specific reactor type that is proposed fall within those of the reactors evaluated at ESP. Subject to adequate demonstration, the determinations rendered at ESP with respect to fuel cycle and transportation environmental impacts are considered resolved for purposes of the COL proceeding.

In accordance with the protocol established for documenting resolution of generic ESP issues, we request that, by reply to this letter, the NRC confirm that the methodology described above and in the attached example is suitable for ESP application environmental reports. To provide for timely resolution of generic issues and continued progress toward submittal of ESP applications in mid-2003, we request that NRC respond by April xx, 2003.

We look forward to your confirmation of the understandings and expectations described above related to ESP-8. If you have any questions concerning this request, please contact me (rls@nei.org or 202-739-8128) or Russ Bell (rjb@nei.org or 202-739-8087).

Sincerely,

DRAFT

Ron Simard

Enclosure

c: Ronaldo V. Jenkins, NRC/NRR
NRC Document Control Desk

Draft ESP- 8 – Attachment 1
February 21, 2003

Examples of the Methodology for ESP Application Environmental Report Preparation for Tables S3 and S4

Environmental impacts are typically measured by impacts to environmental media and health impacts to workers and the public. In the case of the uranium fuel cycle, these impacts manifest themselves as land use, water use, chemical and radioactive emissions to both air and water, and waste disposal. There are also impacts to the worker and the public from the processing and transport of materials from each step of the process. Tables S3 and S4 were developed to provide an estimate of the environmental impacts from the nuclear fuel cycle and transportation. These tables were developed in a generic and non-site specific. The environmental impacts of the nuclear fuel cycle do not depend significantly on the location chosen for the reactor. They depend primarily on the fuel cycle and transportation requirements for materials and services.

The fuel cycle and transportation activities listed below are used to define fuel cycle requirements. The environmental impacts shown in Tables S3 and S4 were determined from these fuel cycle requirements. For each activity, the metric used in determining the fuel cycle requirements is identified. This list was basically taken from NUREG-0116, Table 3.2, and assumes no recycling.

Mining	ore supply in metric tons
Milling	U_3O_8 in metric tons
UF6 production	natural UF_6 in metric tons
Enrichment	enriched UF_6 in metric tons and the separative work units
Fuel Fabrication	fuel loading in metric tons
Reprocessing	not considered
Solid radwaste	Curies from operations
Decontamination & Decommissioning	Curies and cubic meters of LLW
Transportation	Number of shipments for initial core loading
	Number of reload shipments per year
	Fission product inventory in Curies per MTU
	Actinide inventory in Curies per MTU
	Kr-85 activity in Curies per MTU
	Total radioactivity in Curies per MTU
	Decay heat in watts per MTU
	Number of shipments of irradiated fuel
	Heat per cask shipment
	Truck density in trucks per day
	Rail density in cars per month

The approach proposed for evaluating the environmental impacts for the ESP applications is not inconsistent with what has been done previously (e.g., to address burnup limits) and is considered appropriate pending update of Tables S3 and S4 by the NRC. The approach relies on the environmental impacts shown in Tables S3 and S4 and compares the fuel cycle requirements for a range of reactor types to those used to arrive at the values in the current Tables. The purpose of this evaluation is to determine whether the environmental impacts shown in the Tables are bounding for additional reactor types, considering only a uranium fuel cycle only. ESP applicants do not at this time intend to extend this approach to the evaluation of plutonium, mixed oxide and other potential fuel cycles are not part of this approach. For cases where the fuel cycle requirements exceed those used to calculate the environmental impacts in Tables S3 and S4, an evaluation will be performed to determine whether fuel cycle technology improvements [for example, reduced energy consumption for uranium enrichment] mitigate the environmental impact from the increased fuel cycle requirement. If fuel cycle requirements are identified that show increases without an identified mitigating technology improvement, alternatives for estimating the associated environmental impacts will be presented.

Activities and associated requirements for the fuel cycle and transportation process have been obtained from the vendors of the technologies used to establish the Plant Parameter Envelope [PPE] for the ESP applications. These requirements will be compared to the requirements for the reference LWR that formed the basis for Tables S-3 and S-4. Since several of the new reactor technology configurations provide more electricity than the referenced plant, these requirements are normalized as appropriate. Based on the results of these comparisons, there will be one of two outcomes: (1) A fuel cycle requirement associated with the new reactor technologies is less than the referenced plant requirement, therefore the associated impacts shown in Tables S3 and S4 are bounding. (2) A fuel cycle requirement is greater than the referenced plant and further examination is needed, including consideration of mitigating factors that offset increases in environmental impacts implied by the greater fuel cycle requirements.

For example with respect to mining, a key environmental impact is the amount of land that is impacted by the mining technology. If the new reactor technology requires less ore, then the expected amount of land used will be less because WASH 1248 used conservative assumptions on uranium mining techniques compared with current practice. Even if the uranium ore requirements have increased, the Table S3 associated environmental impacts may nonetheless bound the expected values. The "Red Book" *Uranium 2001: Resources, Production and Demand* states that at the end of 2000 the only production in the United States was by *in situ* leaching (ISL). This technology has a very low environmental impact compared with the open-pit and underground mining that was assumed in the WASH 1248 report.

The key activities for the uranium fuel cycle are mining, milling, and conversion to uranium hexafluoride, enrichment, fabrication, waste disposal and transportation. These are the same activities identified in WASH-1248 as the main contributors to any environmental impacts. Reprocessing is not currently being considered and as such is excluded from the ESP fuel cycle analysis. These are the same activities identified in WASH 1248 as the main contributors to any environmental impacts.

The parameters needed to calculate annual fuel cycle requirements, regardless of technology, are the annual fuel loading [mass of uranium] and the enrichment [percent U235]. Annual uranium ore [mass of natural uranium], enrichment services [SWU] and other fuel cycle requirements are determined and compared to the requirements assumed in WASH 1248, "Environmental Survey of the Uranium Fuel Cycle," and NUREG 0116, "Environmental Survey of the Reprocessing and Waste Management Portions of the LWR Fuel Cycle." More specifically if annual reactor fuel uranium mass and enrichment requirements are known, then the quantity of UF_6 required can be determined, the SWU can be calculated, the U_3O_8 that is needed can be specified and lastly the quantity of uranium ore required can be computed.

For the fuel and radioactive waste transportation activity, the key characteristic is the number of shipments. Existing NRC cask design and transportation licensing regulations assure that the other matters of potential concern, such as shielding and transport safety are properly addressed.

Most of the activities and environmental impacts associated with the nuclear fuel cycle and transportation are independent of reactor design. For fuel cycle activities, uranium ore recovery, milling, UF_6 conversion and enrichment are independent of reactor technology. Only fuel fabrication is dependent on reactor design. The only major technology-dependent difference for transportation would be the cask design differences required for various fuel assembly types and source terms. However, all cask designs have to meet NRC licensing regulations.

Conditions for use of Table S-4 [reactor power level, fuel form, enrichment, fuel cladding, burnup, modes of transport, and irradiated fuel decay time prior to shipment] are evaluated by calculating annual fuel cycle requirements to determine whether the Table S-4 environmental impacts are bounding.

In summary, based on the results of the comparison of the above fuel cycle requirements with the reference LWR, the following steps will be taken. If an ESP fuel cycle requirement is less than the reference LWR, then it will be stated that based on the conservative analysis of the WASH reports, the associated environmental impacts shown in Tables S3 and S4 are suitable for use in ESP applications. If an ESP fuel cycle requirement is greater than the reference LWR, then a further analysis will be conducted. If there are other mitigating factors, practices and changes from the original WASH assumptions, then it may be possible to show that the associated environmental impacts are still less than the values in Tables S-3 and S-4. If it can't be shown that the impacts are bounded, alternatives for estimating the associated environmental impacts will be presented.

The following two examples illustrate use of this approach to evaluating the suitability of Tables S3 and S4 to represent the environmental impacts of additional reactor types. ESP applications will present a complete discussion of such evaluations and conclusions for NRC review and approval.

1) Transportation example: The transportation of fuel and waste for the reference LWR required 18 truck shipments for initial core load, 6 reload shipments per year, 60 spent fuel shipments per year, and 46 radwaste shipments per year for a total of 130 truck shipments per year. This value is translated into Table S-4 as less than 1 truck shipment per day. The approach used is to determine the total number of shipments (fuel and LLW) for each of the reactor technologies. If the requirement is less than the reference case, then Table S-4 is appropriate for estimating the associated environmental impacts for the ESP applications.

2) Enrichment services example: The reference LWR required 127 units of enrichment services (MTU of SWU) annually. Based on the annual fuel mass and enrichment requirements provided by the reactor vendors, the annual SWU requirements for the ESP applications are higher than those assumed in the WASH reports. Environmental impacts associated with enrichment services are part of the overall fuel cycle environmental impacts shown in Table S3. However, a close look at the original analysis (WASH 1248) shows the impacts are almost totally from the electrical generation needed for the gaseous diffusion enrichment process. These impacts are the emissions from the electric generation that is assumed to be from coal plants and from the associated water use to cool the plants. A significant fraction of the enrichment services to US utilities today is provided from European facilities using centrifuge technology rather than the fifty-year-old gaseous diffusion technology. Two companies have now announced plans to develop centrifuge technology enrichment in the US. Centrifuge technology requires less than 10 % of the energy needed for the gaseous diffusion process. The gas reactor technology with the highest annual

SWU use requires approximately 30% greater that the SWU assumed in the WASH reports. Therefore, the environmental impacts shown in Table S3 associated with SWU services are still expected to be bounding for the ESP applications. The ESP applications will present a complete discussion of this analysis and conclusion for NRC review and approval. Additionally, if a COL applicant decided to use a design with a high SWU requirement and gaseous diffusion enrichment, the environmental impact of enrichment services would be subject to review at COL.

ESP-9

Title: Criteria for assuring control of the site by the ESP holder

Background: In general, an Early Site Permit does not grant approval to conduct work activities (except in accordance with 10 CFR 52.17(c)); therefore the degree to which an ESP holder "controls" the site is considerably limited when compared to the control which must be demonstrate by a COL applicant.

Industry Approach:

- An ESP can have joint holders.
- If one or both of the ESP holders owns the property that is the subject of the ESP, unless otherwise specified in the ESP application, it is assumed that the ESP holder has sufficient legal rights and authority over the property to carry out the objectives of the ESP, and that it has the authority to ensure the requirements of 10 CFR 52.35 (Use Of The Site For Other Purposes) are satisfied.

- If the ESP holder(s) does not own the property that is the subject of the ESP, the ESP applicant must attest in the application to the fact that the ESP holder has been or will at the appropriate time be granted sufficient legal rights and authority over the property to carry out the objectives of the ESP. Further, the ESP holder will establish the appropriate relationship with the property owner that ensures the requirements of 10 CFR 52.35 (Use Of The Site For Other Purposes) are satisfied.
- For the purposes of this ESP-9 generic issue, sufficient legal rights and authority means that the ESP holder(s), among other things,
    - can make emergency planning agreements pursuant to 10 CFR 52.17(b)(3);
    - will be responsible for the conduct of ESP-authorized pre-COL construction activities pursuant to 10 CFR 50.10(e)(1) and 10 CFR 52.17(c); and
    - will be responsible for the implementation of a redress plan as applicable.

NRC feedback and discussion requested on March 5.

ESP-12 Follow-Up Discussion

The NRC staff's February 12 letter on ESP-12, "NEPA Consideration of Severe Accident Issues," said while SAMAs could be deferred to COL if detailed design information was not available for ESP, the staff expects that ESP applications will address the environmental impacts of severe accidents. The staff letter indicated that this is consistent with the staff position articulated in SECY-91-041.

Based on the staff's feedback, we have reviewed the following generic analyses with respect to their use in addressing severe accident impacts at ESP:

- NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Plants," which provides a detailed evaluation of the accident risks for five U.S. nuclear power plants based on detailed level 1, 2 and 3 PRA studies for these plants and associated sites.
- NUREG/CR-2239, "Technical Guidance for Siting Criteria Development," which evaluates the impact of severe accidents on a generic basis, independent of reactor design, for all 91 reactor sites as of 1982 [including the three ESP application sites].

We are considering an approach that would provide a generic discussion of severe accident impacts in ESP applications based on these NRC generic analyses. For example, NUREG/CR-2239 demonstrated that severe accident environmental impacts were not significantly dependent of site meteorology or emergency planning, but were sensitive to population distribution. Thus, the ESP/ER would focus on demonstrating that the population distribution assumed in the generic analyses remains applicable for the site. Other potential environmental impacts are still under evaluation.

The discussion in ESP applications would demonstrate that severe accident environmental impacts at the proposed site are bounded by the results of the generic analyses based on existing plant designs. The Commission concluded in its 1985 Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants that existing plants pose no undue risk to public health and safety. The Commission Policy Statement also identifies the expectation that new plants will achieve a higher level of severe accident safety performance than existing plants. This expectation has been realized in each of the three standard designs certified by the NRC to date. Thus severe accident impact evaluations for ESP based on existing plant designs will bound those for future plant designs that are expected to satisfy the Commission's Severe Accident Policy Statement.

At COL, for an applicant referencing a certified design and an ESP, severe accident impacts would, absent a significant new environmental issue associated with the design or site, be considered resolved for purposes of the COL proceeding.

We note that draft RS-002 indicates that ESRP Section 7.2 is applicable to the ESP review. However, portions of Section 7.2, (e.g., Review Interfaces) require use of detailed design information that may not be available at time of ESP, as well as consideration of SAMAs, a topic that may be deferred to COL.

Please provide your feedback on our interpretation of your Feb. 12 letter and the approach outlined above.

Primary source documents for evaluating severe accident risks for Early Site Permit applications:

1. NUREG/CR-2239, "Technical Guidance for Siting Criteria Development", Sandia National Laboratories, December 1982.
2. NUREG-1150, "Severe Accident Risks: An Assessment for Five Nuclear Power Plants," U.S. Nuclear Regulatory Commission, June 1989.

Quotes from Reference 1:

- The work has been primarily focused toward the development of generic siting criteria, uncoupled from specific plant design. To achieve this end, the NRC staff developed a representative set of severe accident release source terms which covers the full spectrum of postulated accident releases for typical light water reactors.
- The NRC staff would assign typical probability values to the source terms for a range of light water reactor designs as follows [see table below for SST definations]:
    - Probability of SST1 release 1×10^{-5} / reactor year
    - Probability of SST2 release 2×10^{-5} / reactor year
    - Probability of SST3 release 1×10^{-4} / reactor year
- The siting source terms were used to calculate accident consequences at 91 U.S. reactor sites using site specific meteorology and population data and assuming an 1120 MWe reactor.
- Contained in this report are sensitivity studies for the major parameters important to siting decision making.
- Given the source term assumptions, large consequences are calculated. However, the risks (probabilities times consequences) posed by such accidents are very small.
- Summary
    - Estimates of the number of early fatalities are very sensitive to source term magnitude.
    - Mean health effects (average result for many weather sequences) are relatively insensitive to meteorology.
    - Calculated consequences are very sensitive to site population distribution.
    - Early fatalities and early injuries can be significantly reduced by emergency response actions.
    - Population densities (people/sq mi) about the 91 sites have the following maximum, 90th percentile and median values within the indicated distance intervals:

Distance (mi)	0-5	0-10	0-20
Maximum	790	660	710
90th percentile	190	230	380
Median	40	70	90

e focus on In the The focus of ER discussions apt because these evaluations are largely independent of indicates that the evaluation of severe accident impacts is These analyses indicate performed The overall recommended approach for ESP applications is to develop a generic section [to be identical in all three applications] on severe accidents. This generic approach would be based on NUREG/CR-2239 and NUREG-1150. Some of the results of studies for license renewal may also be relevant especially the GEIS section on severe accidents.

The severe accident section would compare the population distributions for the three sites with the NUREG/CR-2239 table shown below. NUREG/CR-2239 shows that accident consequences are not significantly dependent on site meteorology and thus no detailed discussion of meteorology would be presented. The discussion would emphasize that the results from NUREG/CR-2239 are independent of reactor design – with special emphasis that the results assume failure of all safety systems including containment. The section would conclude that the environmental impact of severe accidents is small on a generic basis independent of design. The only site characteristic important to confirm this generic conclusion is population density. The study shows that emergency planning is important but considers cases where no protective measures are taken.

The proposed ER 7.2 would then show that the combination of accident probabilities combined with source term for the ABWR, AP600 and AP1000 are substantially less than those assumed for both NUREG/CR-2239 and NUREG-1150. ER 7.2 would use NRC's own language to indicate that other advanced designs would have to meet safety goal criteria and exhibit low public risk or would not be licensed by NRC. At COL the selected design would be shown to have severe accident environmental impacts that are not significantly greater than shown at ESP. During design certification or at COL for a non-certified design, potential severe accident design mitigation analyses would be performed. Some additional evaluation of SAMA [for example procedural rather than design alternatives] may be required for COL.

- Brief Descriptions of Characterizing the Accident Groups Within the NRC "Accident Spectrum"

Group 1	Severe core damage. Essentially involves loss of all installed safety features. Severe direct breach of containment.
Group 2	Severe core damage. Containment fails to isolate. Fission product release mitigating systems (e.g., sprays, suppression pool, fan coolers) operate to reduce release.
Group 3	Severe core damage. Containment fails to by basemat melt-through. All other release mitigating systems function as designed.

Observations on NUREG/CR-2239

1. This NUREG makes **extremely** conservative source term and accident probability assumptions and still concludes that risks from severe accidents is small.
2. Research and evaluations since NUREG/CR-2239 support substantial reductions in both expected source term and accident probability [NUREG-1150, individual plant PRA evaluations, and development of the alternative source term.]
3. While NUREG/CR-2239 states that it is specific to light water reactors, the accident source terms and probabilities would certainly be expected to bound any future design that NRC would license. Groups 1 through 3 source terms assume severe core damage combined with containment failure. Future reactor designs that rely on preventing core damage by design rather than providing containment of the damaged core would certainly still have to prove to NRC that the probability of core damage was sufficiently low to limit severe accident consequences to less than that assumed by NUREG/CR-2239. Reactors with power levels higher than the 1120 MWe assumed would have a small impact on consequences.
4. Severe accidents were shown to have little dependence on site meteorology. The ESP applications should therefore be able to handle this in a generic manner.
5. The 91 sites evaluated were all sites licensed in 1982 including Clinton, North Anna and Grand Gulf [as shown in Appendix A of the report].
6. While population distribution was an important factor, I would guess that the three ESP sites would likely be less than the 90th percentile site. Note that the report did not conclude that any exiting site population value was unacceptable. It is also interesting to note that even the maximum site in 1982 met the 1,000 person per square mile criteria.

Outline of ESP-13 status update for discussion at 3/5 meeting

Geotechnical Work

This work involves the field explorations, laboratory testing, and preliminary engineering evaluations.

- The field work is complete. This work included the drilling and sampling, the cone penetrometer testing, the geophysical logging, and the surveying. Reports for this work have also been completed.

- The laboratory standard testing is complete. Results from the standard testing program have been provided and this activity is complete.

- The laboratory dynamic testing is complete. Data has been received; however, the report is not complete.

- A comparative study of conditions at the ESP site versus the existing site has been completed (conditions are essentially the same), and we have completed our preliminary assessments of liquefaction potential, bearing capacity, settlement, and earth pressures. Since the sites compare so closely, we are relying on information in the existing facility Updated Safety Analysis Report for the bearing capacity, settlement, and earth pressure conditions. A new liquefaction analysis was completed to account for changes in the methods of analysis.

- A draft of the Geotechnical Report describing the three preceding tasks and their results is being finalized. The key information in the Geotechnical Report will be presented in Section 2.5 of the Site Safety Analysis Report (SSAR) with the full Geotechnical Report included as an appendix.

Seismic Hazards Work

This work involves the updating of information in the seismic hazards study conducted by EPRI in the mid-to-late 1980s.

- The sensitivity studies for the site were completed last Fall. These studies concluded an update of the probabilistic seismic hazards analysis (PSHA) was needed for the Exelon ESP Site.

- The background work is nearly complete. This background work involved updates on the seismic source mechanisms and possible levels of shaking, and the paleo-liquefaction studies that were carried out for the area. These activities are complete. This background work also includes the ground motion modeling update being coordinated by EPRI. The final SSHAC Level III workshop is planned for March 11th to reach consensus on the ground motion models that are appropriate for eastern and central US.

- The PSHA update is nearly complete (waiting for the EPRI ground motion models and weights to estimate the hazard). The final step of the PSHA will involve conducting the analysis using an updated EPRI hazards code.

- The key information in the PSHA will be presented in Section 2.5 of the SSAR with the full PSHA included as an appendix.

- Notice: The preliminary seismic information indicates possible exceedance of the standard plant seismic design basis spectrum at frequencies above about 10 Hz. At least one of the ESP applicants intends to utilize the methodology developed in EPRI TR-102470 to reduce the impact of high-frequency motions.

Concluding Comments

An overall wrap-up on the seismic hazard demonstration will likely not be possible until late April or early May. Thus, by the time the NRC Staff could evaluate the presented information and provide any meaningful feedback, the first of the applications will be submitted.

February 21, 2003

<u>Industry Comments on IMC-2501</u>
<u>Nuclear Reactor Inspection Guidance, ESP Phase (10/08/02)</u>

Per the NRC staff's request during our Jan. 29, 2003, public meeting, we are providing specific comments on IMC-2501 for discussion at our March 5 public meeting and as input to staff revision of the document.

IMC-2501 currently contains a number of statements regarding QA requirements for ESP. We understand that IMC-2501, in particular, Section 2501-05.05, will be revised to reflect the following staff views: (1) that the staff does not hold that ESP applicants are required to have an Appendix B program, (2) that QA program descriptions are not required to be included in ESP applications or reviewed by the NRC, and (3) that Appendix B will be used by NRC staff as a guide for assessing the quality of site safety analysis information.

In revising IMC-2501, we strongly recommend the staff avoid use of the language from its Feb. 3 letter on ESP-3 to the effect that quality controls applied to ESP activities associated with site safety should be "equivalent in substance" to the controls described in Appendix B. This language is at best confusing and at worst in conflict with the staff's position that an Appendix B program is not required for ESP. Prior to the Feb. 3 letter, both we and the NRC staff had used alternative language to describe QA requirements and expectations for ESP. First, in its Dec. 5 position statement, the staff said that they intend to "asses the ESP applicant's QA program to ensure that the appropriate QA elements are in place in order to (1) to establish a baseline for future use during the COL process, and (2) to assess any potential impacts on the staff's findings." In its Feb. 3 letter, the staff clarified that the phrase "'baseline for use' refers to the need for the staff to determine that QA measures applied to information submitted for review at the ESP stage are adequate, such that the staff can accept the use of this information, as embodied in an ESP, in support of a later CP, OL or COL application." The industry agrees with this description. Similarly, the NRC's Feb. 3 letter indicated agreement with the industry's description in our Dec. 20, 2002, ESP-3 resolution letter. Item seven of the NEI letter stated that, "Because of the finality of the issues resolved as part of the ESP process, the staff must have confidence in the site safety analysis information in order to make its conclusions. It is expected that the NRC staff will review the applicant's quality processes and sources of information to develop the necessary confidence in ESP information."

Both of these descriptions have been accepted by both the industry and the NRC, and both are preferable to describe the nature of NRC staff QA reviews of ESP application information. Both the earlier NRC and industry descriptions are clearer and thus preferable to the "equivalent in substance" language in the staff's Feb. 3 letter because the earlier descriptions place the focus of NRC review on the appropriateness of the analysis methodology, data sources and results, not on whether quality processes used are equivalent to those of Appendix B.

ESP-19

Title: Addressing effects of potential new units at an existing site

Background: If an Early Site Permit applicant proposes to approve a site for future nuclear units upon which an operating nuclear facility is co-located, interface issues involving the proposed and operating units will exist and will need to be addressed through the appropriate processes.

Industry Approach:

- The operating unit licensee is responsible for and has authority over the "owner-controlled-area."
- The ESP holder is responsible for compliance with 10CFR52.35 (Use of the site for other purposes).
- If the ESP holder and the operating unit licensee are different entities, appropriate managerial and administrative controls would need to be established to ensure compliance with 10CFR52.35.
- 10CFR50.34(a)(11) states "On or after February 5, 1979, applicants who apply for construction permits for nuclear power plants to be built on multiunit sites shall identify potential hazards to the structures, systems and components important to safety of operating nuclear facilities from construction activities. The provisions of 10CFR50.34(a)(11) are not invoked by 10CFR52.17(a)(1); therefore the requirements of 10CFR50.34(a)(11) are not applicable to the ESP application. The requirements of 10CFR50.34(a)(11) are applicable to a COL application.
- If an ESP applicant proposes to perform activities in accordance with 10 CFR 52.17(c), the applicant, at its discretion, may include in the ESP application a discussion of administrative controls similar to the discussion required by 10 CFR 50.34(a)(11).
    - The operating unit licensee is required pursuant to its license to evaluate and appropriately control 10 CFR 52.17(c) activities within its owner-controlled-area, which represent potential hazards to the structures, systems and components of the operating unit. In this regard, 10 CFR 50.59 is the controlling regulation for the operating unit.
    - Impacts from 10 CFR 52.17(c) activities would also be evaluated by the operating unit licensee against its programs (e.g., emergency, security, environmental protection, and decommissioning plans) in accordance with the regulations applicable to the operating unit. None of the evaluations performed by the operating unit licensee are part of the ESP application.
    - Evaluations resulting from 10 CFR 52.17(c) activities would be conducted by the operating unit licensee prior to commencing any physical work authorized pursuant to the ESP or other NRC granted work authorization.

NRC feedback and discussion requested for March 5

6. Section 2501-05, item 05.05.a, states "the applicable criteria of 10 CFR 50 Appendix B are those criteria which can directly relate to the pedigree or genesis of any safety-related or risk-significant structure, system, or component (SSC)."

The stated purpose of Appendix B is to establish "quality assurance requirements for the design, construction, and operation of those structures, systems, and components. The pertinent requirements of this appendix apply to all activities affecting the safety-related functions of those structures, systems, and components; these activities include designing, purchasing, fabricating, handling, shipping, storing, cleaning, erecting, installing, inspecting, testing, operating, maintaining, repairing, refueling, and modifying." The identified guidance inappropriately expands the scope of Appendix B to "risk-significant SSCs" and anything related to the "pedigree or genesis of any safety-related SSC."

7. Section 2501-05, item 05.06, discusses a Limited Work Authorization that might be issued under §50.10(e)(1) and states "This may include extension of previously permitted activities subject to 10 CFR Part 50, Appendix B, such as the continuance of site exploration..."

The indication that site exploration activities are subject to Appendix B is not consistent with the statements in 10 CFR Part 100, 100.23(b) and Appendix A (section II) which indicate the geological, seismological, and engineering characteristics investigations required by 100.23(c) and Appendix A are "within the scope of investigations permitted by §50.10(c)(1)." The activities permitted by §50.10(c) are identified therein as "other pre-construction monitoring to establish background information related to the suitability of the site or to the protection of environmental values." These activities have not previously been considered to be within the scope of Appendix B activities.

8. We agree with Section 2501-05, item 05.05, where it states: "the quality and pedigree associated with those parts of the ESP application not applicable to Appendix B will be reviewed to recognized industry codes and standards."

February 21, 2003

Other Comments on IMC-2501

1. Section 2501-01, PURPOSE, states: "...The ESP phase is implemented when the NRC receives formal notification under 10 CFR Part 52 of an applicant's intention to apply for an ESP."

The NRC may not receive formal notification of an applicant's intent to apply for an ESP since there is no requirement for applicants to notify NRC of such intent. It is desirable, a good practice, policy, etc., but there is no requirement. The ESP phase should begin when the NRC either receives formal notice of intent to apply for an ESP or receives an application.

2. Section 2501-01, PURPOSE, states: "...It continues until the ESP expires after 20 years or a combined operating license or construction permit is issued."

The statement assumes the ESP applicant requests a 20-year ESP. The phrase "after 20 years" should be deleted since the ESP could be for a period as short as 10 years.

3. Section 2501-03, DEFINITIONS, defines "Quality Assurance Program/QA Commitments" in 03.10 as the information required by 10 CFR 50.34(a)(7).

Because a QA program description is not a requirement for ESP, this definition should be revised as follows: The terms QA Program and QA Commitments relate to the quality processes and controls implemented by the ESP applicant.

4. Section 2501-05, item 05.02, states "the application will be reviewed according to 10CFR Part 50 Appendix B, as required by 10CFR Part 52.18."

To be correct, the following should be added to the end of this statement: "if the QA program or description thereof is included in the application and the applicant has committed to using App. B for at least a portion of the application." As discussed above, NRC review in accordance with Appendix B is not a requirement for ESP.

5. Section 2501-05, item 05.03, states "ESP Phase Inspection Guidance, Enclosure 1 to IMC-2501, provides guidance which may be applicable during inspections, audits, or site visits."

None of the guidance documents in Enclosure 1 to IMC-2501 are currently available for review or use. We understand that the referenced guidance are currently being developed and will be available to ESP applicants and the public prior to their first use by the NRC.

NRC FORM 659
(5-2002)

U.S. NUCLEAR REGULATORY COMMISSION

Category
2

NRC PUBLIC MEETING FEEDBACK

Meeting Date: 03/05/2003

Meeting Title: MEETING WITH NUCLEAR ENERGY INSTITUTE (NEI) REGARDING EARLY SITE PERMIT APPLICATIONS

In order to better serve the public, we need to hear from the meeting participants. Please take a few minutes to fill out this feedback form and return it to NRC.

1. How did you hear about this meeting?

☐ NRC Web Page ☐ NRC Mailing List ☐ Newspaper
☐ Radio/TV ☐ Other ______

		Yes	No	Somewhat
			(Please explain below)	
2.	Were you able to find supporting information prior to the meeting?	☐	☐	☐
3.	Did the meeting achieve its stated purpose?	☐	☐	☐
4.	Has this meeting helped you with your understanding of the topic?	☐	☐	☐
5.	Were the meeting starting time, duration, and location reasonably convenient?	☐	☐	☐
6.	Were you given sufficient opportunity to ask questions or express your views?	☐	☐	☐
7.	Are you satisfied overall with the NRC staff who participated in the meeting?	☐	☐	☐

COMMENTS OR SUGGESTIONS:

Thank you for answering these questions.

Continue Comments on the reverse.

OPTIONAL

Name ______ Organization ______

Telephone No. ______ E-Mail ______ ☐ *Check here if you would like a member of NRC staff to contact you.*

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WASHINGTON DC 20277-2904

Agenda
March 5, 2003 Meeting
Nuclear Energy Institute (NEI)

9:00 a.m.	Introductory Comments		
	Follow-up Items from January 29, 2003 Meeting		
	Follow-up Discussion on NRC responses to NEI Issue Resolution Letters		
	ESP-2:	ESP Inspection Guidance	
	ESP-4:	NRC nominal review timeline	
	ESP-13:	Guidance for ESP seismic evaluations	
	ESP-21:	Form and Content of an ESP	
10:50 a.m.	Break		
11:00 a.m.	[Continue morning item discussions]		
	Opportunity for public comment		
12:00 Noon	Lunch		
1:00 p.m.	[Continue morning item discussions as necessary]		
	ESP-9:	Criteria for assuring control of the site by ESP Holder	
	ESP-19:	Addressing effects of potential new units at an existing site	
	ESP-21:	Understanding the interface of ESP with the COL process	
	Topics for the future meetings		
	Opportunity for public comment		
	Summary		
5:00 p.m.	Adjourn		

NOTE: Specific topics and associated discussion times may change without notice

Contact:
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301-415-2985, rvj@nrc.gov

ESP-12 Follow-Up Discussion

The NRC staff's February 12 letter on ESP-12, "NEPA Consideration of Severe Accident Issues," said while SAMAs could be deferred to COL if detailed design information was not available for ESP, the staff expects that ESP applications will address the environmental impacts of severe accidents. The staff letter indicated that this is consistent with the staff position articulated in SECY-91-041.

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