Questions Concerning Licensing Actions

QUESTION 1. What Commission or staff procedures govern the establishment of scheduled dates for completing action on ESP, COL, or design certification applications?

ANSWER

Detailed schedule models for new reactor reviews are developed 1-2 years in advance of receipt of the actual application. These models are based on experience with similar products, reflecting the best information available to the staff regarding the number and scope of technical issues to be evaluated. The licensing projects are planned, scheduled, and managed using internal administrative procedures and review guidance, such as the Standard Review Plan, to guide the activities of the project team and to inform the technical reviews. To prepare for the upcoming licensing activities, the NRC is developing internal procedures specific to Part 52 licensing activities. However, the overall quality and completeness of the application can have a substantial impact on the developed schedule. Therefore, the staff does not commit to a specific schedule until it has examined the license application and accepted it for docketing.

With respect to hearings on ESP and COL applications, the Commission recently amended its Rules of Practice for adjudications in 10 CFR Part 2 to set model milestones for the conduct of contested proceedings. The Commission included these model milestones in the regulations in 10 CFR Part 2, Appendix B. The regulations also provide that the presiding officer should use the milestones as a starting point, make appropriate modifications to the milestones, and set detailed litigation schedules based upon all relevant information.

QUESTION 2. What Commission or staff procedures apply to making changes in established schedules?

ANSWER

Once the staff issues a project schedule, each activity is tracked at a program, division, office, and Commission level to ensure timely completion of all supporting activities. Schedules are developed based upon assumptions, including the scope and nature of technical issues to be addressed and the applicant's commitments to supporting the review with timely and complete information. Schedule changes may be required when an application does not meet these assumptions. Examples of circumstances which can require schedule changes include:

- Identification of a significant safety issue that needs additional confirmatory review,
- Unanticipated changes in scope of work,
- Inadequate or incomplete applications,
- Inadequate responses to staff questions on safety issues,
- Applicant failure to meet schedular commitments for submittal of supporting documentation, including responses to staff questions on safety issues,

- Applicant changes to the project made after the NRC staff has completed the bulk of its review

The NRC can take action to reallocate resources to address schedule challenges directly associated with its review activities, and can often make adjustments to preserve schedule commitments. This flexibility is considerably reduced when the applicant does not provide adequate or timely information.

QUESTION 3. Does the Commission view these schedules as commitments and performance objectives, or can they be changed simply as circumstances and managers' judgments dictate?

ANSWER

The NRC incorporates milestones supporting these schedules into its internal performance monitoring and documents its overall performance in external reports to the public and Congress.

Once a schedule for a project has been issued to an applicant and published on the NRC's website, all milestones and supporting tasks are tracked on a "real time" basis to ensure completion of the activity. Changes to the published milestones are made for the reasons stated in the response to question #2 or after careful consideration of the specific circumstances surrounding a potential schedule delay. Schedule changes for major projects such as an ESP, COL, or design certification are made only after senior management review.

QUESTION 4. What measures are the Commission taking to ensure the agency better anticipates and manages its workload, and keeps established commitments?

ANSWER:

The NRC is applying lessons learned from its successful reviews of design certification applications (most recently the AP1000) and license renewals in its planning for upcoming new reactor licensing activities. These efforts demonstrate the NRC's ability to apply its planning, budgeting, and performance monitoring processes to complete large and complex reviews on schedule.

The NRC's planning for new reactor licensing relies upon information provided by prospective applicants regarding their plans for new reactor deployment. This information is essential so that the staff can realistically project the number of applications and anticipate their scope, considering factors such as the reactor technology, site-specific characteristics, and licensing processes to be used.

Based on current information, the NRC is actively preparing to begin review of 11 or more new reactor license applications, two or more design certifications, and one or more Early Site Permits in FY 2007 and FY 2008. Based on this scope of work, the NRC has acted quickly to realign the organization, establishing a New Reactor Licensing Division that includes dedicated staff to plan for and manage the new reactor licensing projects. Staff activities include development of review guidance and other technical infrastructure, and detailed project planning to identify needed resources and technical skills. The agency has an aggressive hiring strategy and plans to hire more than 350 new employees to support the technical and legal review of these applications.

QUESTION 5. What is the Commission's goal for the length of time it takes to review ESP applications and issue the permits?

ANSWER

The NRC is assessing lessons learned from review of the first three ESP applications to determine whether improvements can be made to the existing nominal 21 month schedule for completion of the final safety evaluation report (FSER) and final environmental impact statement (FEIS). As always, the staff's ability to complete the reviews in a timely manner is highly dependent on the quality and completeness of the applications.

After completion of the FSER and EIS, a mandatory hearing, required by Section 189 of the Atomic Energy Act of 1954, as amended, is held by an Atomic Safety and Licensing Board to review the staff's findings. In completing the hearing phase the Board would also consider any contentions raised by affected stakeholders that have not previously been resolved. On average, completion of the hearing phase is expected to take 12 months and, if the finding is in favor of the applicant, the NRC can then issue the ESP. The NRC expects to complete the hearing phase for both Grand Gulf and Clinton in less than 12 months.

QUESTION 6. What is the Commission's goal for the length of time it takes to review COL applications and issue the licenses?

ANSWER

Using the NRC's current review approach, it is expected that a license for a single new plant can be issued in 42 months, including 30 months for the application review and 12 months for completion of the hearing phase, including the mandatory hearing. This review estimate presumes that the applicant uses an Early Site Permit, references and adheres fully to a certified design, and submits an adequate application that contains all the necessary information to complete the review. The NRC is evaluating the current review approach to determine if efficiencies in resource needs and schedule could be achieved in NRC's review of COL applications while maintaining the requisite safety review.

QUESTION 7. What is the Commission's goal for the length of time it takes to review design certification applications and issue the certificates?

ANSWER

The NRC estimates that a design certification application review will take 42-60 months, depending on the extent to which the design differs from those previously reviewed, whether there is a need for testing and the extent of the testing program, and whether policy matters need to be addressed. The schedule is also highly dependent on the quality and completeness of the application and the applicant's ability to provide timely and complete information as the staff identifies issues in the course of the review. The NRC is developing plans to align dedicated staff resources to keep NRC's component of the overall schedule as short as possible.

QUESTION 8. Please provide a list of the 25 plants most recently authorized to operate with the corresponding dates that license applications were filed and licenses issued authorizing commencement of operations.

ANSWER

Enclosure 2 lists the 25 plants most recently authorized to operate. These plants were licensed under the 10 CFR Part 50 process, in which the applicant applied first for a construction permit and then several years later for an operating license.

The Part 50 process for licensing a plant requires that applicants apply for and be granted a construction permit prior to beginning actual construction. The application for a construction permit includes information necessary to complete the review relative to site safety, emergency planning, and environmental impact, but historically included only preliminary information related to the design of the plant. After the construction permit had been granted, the applicant would complete the design of the plant, which would be submitted to the NRC along with operational program information in the operating license application. The reviews by the NRC and licensing boards were completed, in all instances, before completion of construction, but the NRC could not issue the operating license until the construction of the facility had been substantially completed commensurate with the power level for which operation was sought. With respect to the plants listed in Enclosure 2, the issuance of the operating license occurred many years after the submission of the operating license application for a variety of reasons, including, for example, decisions by applicant's management based on economic factors and delays in the completion of construction as a result of problems with management of design and quality assurance/quality control break-downs during construction.

QUESTION 9. Please provide a list of requirements that constitute a complete design certification application.

ANSWER

The requirements for the contents of applications for design certification are included in 10 CFR 52.47. These requirements are summarized below.

All applications for design certification must contain the following:

- Technical information required for construction permits and operating licenses, and which is technically relevant to the design and not site-specific;

- Demonstration of compliance with any technically relevant portions of the "Three Mile Island requirements," which are set forth in 10 CFR 50.34(f), with a few noted exceptions;

- Site parameters postulated for the design, and an analysis and evaluation of the design in terms of such parameters;

- Proposed technical resolution of those Unresolved Safety Issues and medium- and high-priority Generic Safety Issues that are identified in the version of NUREG-0933 current on the date six months prior to the application and which are technically relevant to the design;

- Design-specific probabilistic risk assessment;

- Proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that a plant which references the design is built and will operate in accordance with the design certification;

- The interface requirements to be met by those portions of the plant for which the application does not seek certification;

- Justification that compliance with the interface requirements discussed above is verifiable through ITAAC; and

- A representative conceptual design for those portions of the plant for which the application does not seek certification.

The application must contain a level of design information sufficient to enable the Commission to judge the applicant's proposed means of assuring that construction conforms to the design and to reach a final conclusion on all safety questions associated with the design before certification is granted. The information submitted must include performance requirements and design information sufficiently detailed to permit the preparation of acceptance and inspection requirements by the NRC, and procurement specifications and construction and installation specifications by an applicant.

For a standard design that is an evolutionary change from light water reactor designs of plants that have been licensed and in commercial operation, 10 CFR 52.47 requires that the application for certification provide an essentially complete nuclear power plant design except for site-specific elements, such as the service water intake structure and the ultimate heat sink.

For a standard design that differs significantly from light water reactor designs, or a design that utilizes simplified, inherent, passive, or other innovative means to accomplish its safety functions, 10 CFR 52.47 requires that certification be granted only if: (1) the performance of each safety feature of the design has been demonstrated though either analysis, appropriate test programs, experience, or a combination thereof; (2) interdependent effects among the safety features of the design have been found acceptable by analysis, appropriate test programs, experience, or a combination thereof; (3) sufficient data exist on the safety features of the design to assess the analytical tools used for safety analyses over a sufficient range of normal operating conditions, transient conditions, and specified accident sequences, including equilibrium core conditions; and (4) the scope of design is complete except for site-specific elements, such as the service water intake structure and the ultimate heat sink. Alternatively, the applicant could demonstrate that there has been acceptable testing of an appropriately sited, full-size, prototype of the design over a sufficient range of normal operating conditions, transient conditions, and specified accident sequences, including equilibrium core conditions. For a standard design of the type described in this paragraph for which the application does not meet the criterion addressing scope of the design, 10 CFR 52.47 requires that the testing of the prototype must demonstrate that the non-certified portion of the plant cannot significantly affect the safe operation of the plant. The application for final design approval of a standard design of the type described in this paragraph must also propose the specific testing necessary to support certification of the design.

For a modular design, 10 CFR 52.47 requires an application for certification to describe the various options for the configuration of the plant and site, including variations in, or sharing of, common systems, interface requirements, and system interactions. The final safety analysis and the probabilistic risk assessment should also account for differences among the various options, including any restrictions that will be necessary during the construction and startup of a given module to ensure the safe operation of any module already operating.

Once an application is submitted, the NRC must perform an acceptance review of the application, using a checklist similar to the one provided in Enclosure 3 to ensure that the application is complete and acceptable for docketing in accordance with 10 CFR 2.101(a)(2). After the application is accepted and docketed, the NRC will develop the schedule for the review of that application, the duration of which is largely dependent on the quality and level of detail provided in the application which addresses how the proposed design will comply with NRC regulations.

QUESTION 10. Please provide a list of requirements that constitute a complete COL application.

ANSWER

A combined license (COL) application may, but need not, reference a standard design certification (DC) or an Early Site Permit (ESP), or both.

The contents of a complete COL application, as specified in 10 CFR 52.79 include the following:

- If the application references an ESP, the application does not need to contain information or analyses previously submitted to the NRC in connection with the ESP, but must contain information sufficient to demonstrate that the design of the facility falls within the parameters specified in the ESP and to resolve any other significant issues not considered in any previous proceeding on the site or the design.

- If the application does not reference an ESP, the applicant shall comply with the requirements of 10 CFR 50.30(f) by including with the application an environmental report prepared in accordance with the provisions of subpart A of 10 CFR Part 51.

- If the application does not reference an ESP that contains a site redress plan as described in 10 CFR 52.17(c) and if the applicant wishes to be able to perform activities at the site allowed by 10 CFR 50.10(e)(1), then the application must contain the information required by 10 CFR 52.17(c) (i.e., a site redress plan).

- The application must contain the technically relevant information required of applicants for an operating license by 10 CFR 50.34. The final safety analysis report and other required information may incorporate by reference the final safety analysis report for a certified standard design. An application referencing a certified design must describe those portions of the design that are site-specific, such as service water intake structure and the ultimate heat sink. An application referencing a certified design must demonstrate compliance with the interface requirements and have available for audit procurement specifications and construction and installation specifications in accordance with 10 CFR 52.47(a)(2). If the application does not reference a certified design, the application must comply with the requirements of 10 CFR 52.47(a)(2) for the level of design information and shall contain the technical information required by 10 CFR 52.47(a)(1)(i), (ii), (iv), and (v), and (3), and, if the design is modular, 10 CFR 52.47(b)(3).

- The application must include the proposed inspections, test, and analyses, including those applicable to emergency planning, that the licensee shall perform and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations. Where the application references a certified design, the inspections, tests, analyses, and acceptance criteria contained in the certified design must apply to those portions of the facility design that are covered by the design certification.

- The application must contain emergency plans that provide reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency at the site.

- If the application references an ESP, the application may incorporate by reference emergency plans or major features of emergency plans approved in connection with the issuance of the ESP.

- If the application does not reference an ESP, or if no emergency plans were approved in connection with the issuance of the ESP, the applicant shall make good faith efforts to obtain certifications from the local and State governmental agencies with emergency planning responsibilities that the proposed plans are practicable, that the agencies are committed to participate in future plan development, and that these agencies are committed to executing their responsibilities under the plans in the event of an emergency. The application must contain any certifications that have been obtained. If these certifications cannot be obtained, the application must contain information, including a utility's plan, sufficient to show that the proposed plans nonetheless provide reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency at the site.

To communicate our expectations better with all stakeholders, the staff is preparing a checklist, similar to the one prepared for design certifications (Enclosure 3) for COL applications. The staff expects this checklist to be available by March 2006.

QUESTION 11. Please provide a list of all federal, state and local government authorizations, permits, or other approvals upon which a Construction and Operation License is contingent.

ANSWER

A representative list of the authorization, permits, certifications, and consultations required for activities related to site preparation, construction, and operation of potential new nuclear units (in this case, for the North Anna ESP) is given in Enclosure 4. A list (and the status of such authorizations, etc.) is required to be submitted by an applicant in its Environmental Report by 10 CFR 51.45(d). Actual requirements vary to some extent dependent upon the site, design, and jurisdiction where the proposed facility is located.

QUESTION 12. Of the 350 new employees you intend to hire, how many of [them] will be permanently assigned to new reactor licensing positions? Where will the remaining new employees be placed?

ANSWER

The 350 new employees will be distributed throughout the NRC in support of New Reactor Licensing Activities, with 275 providing direct licensing review support and the remaining new employees providing other support functions. Not all of those hired for direct support will be assigned exclusively to the highly complex new reactor licensing reviews, but assigning those personnel to other work will make more experienced staff available for new reactor work. As a result, a total of 275 new and more experienced staff will be assigned specifically to new reactor licensing reviews.

Questions Concerning the ESP Delays

QUESTION 13. What is the Commission's current schedule for completing actions on the ESP applications?

ANSWER

The dates below are the targeted completion dates:

For the Grand Gulf ESP:

Final Environmental Impact Statement (EIS)	April 14, 2006
Atomic Safety and Licensing Board (ASLB) initial decision	September 2006
Commission decision	January 2007

For the Clinton ESP

Final Safety Evaluation Report (SER) issued	February 17, 2006
Final EIS	July 28, 2006
ASLB initial decision	January 2007
Commission decision	May 2007

Efforts to shorten these schedules at both the ASLB and Commission stages can be expected.

For the North Anna ESP

On January 13, 2006, the applicant submitted a stand-alone supplement to its ESP application to modify the cooling water system design for the potential nuclear reactor at the North Anna site. This supplement also includes a change to allow an increase in the power level of a plant to be located at the North Anna site. The proposed increase in power level affects a number of previously reviewed safety and environmental matters. Since there is a substantial change to the normal cooling design and since the changes proposed by the applicant impact many sections of the application, the staff plans to issue a supplement to the previously issued draft EIS and a supplement to the final SER.

The NRC performed an initial review of the the applicant's January 13, 2006 supplement and determined that several key areas are lacking information necessary for the NRC to complete the necessary review. The NRC informed the applicant of this determination by letter dated February 10, 2006 (Enclosure 5). By the same letter, the NRC provided a schedule for the review of the North Anna ESP showing that the NRC expects to issue the supplemental final SER and the final EIS for the application within 260 days from the date that the applicant addresses the identified deficiencies.

<u>QUESTION 14.</u> How many public comments were received on the ESP applications? How many did the Commission expect to receive?

<u>ANSWER</u>

The staff received over 13,000 comments from 1349 individuals on the North Anna ESP draft environmental impact statement. The staff had anticipated to receive about 500 comments, based on experience with environmental reviews supporting license renewal applications.

The staff received 1570 comments on the Clinton ESP draft EIS, and 3201 comments on the Grand Gulf ESP draft EIS. In all cases, each comment is reviewed to determine whether changes to the EIS are warranted.

<u>QUESTION 15.</u> What actions has the NRC taken to determine why the NRC staff was not prepared for the volume of comments?

<u>ANSWER</u>

The staff has determined that it did not fully appreciate the effect that the Internet could have on the level of stakeholder participation. Since the EISs could be viewed by a much wider audience and since it is acceptable to submit comments by e-mail, many more comments on the ESP EISs were received. These comments originated from locations across the U.S. and some foreign countries.

QUESTION 16. When notified of the need for delay, what specific actions did the Commission take, and what specific direction was given to the staff?

ANSWER

When informed of the need for a delay, the Commission required the staff to explain the cause for the delay, why the problem had not been foreseen, and options to reduce the impacts of the delay. As discussed above, the principal cause of the delay was a significantly higher level of stakeholder participation than the staff had anticipated.

In response to identifying the need for the delay, the Commission and NRC management directed the staff to shift work unrelated to the ESPs to alternate contractors and continue to increase the level of staff involvement by shifting priorities. This allowed the contractor to use additional resources to respond to the comments and to increase the level of staff interactions. In addition, the Commission directed the staff to develop options to minimize the effects of the delay on the program. For example, the staff realigned the order of review for the second and third ESPs because technical issues with the content of the second application called for additional work by the applicant. The Commission and NRC management also directed the staff to develop different tools to manage the processing of public comments for future reviews. The latter effort is part of a larger initiative that has been undertaken by the NRC to prepare for future new reactor work.

QUESTION 17. Did the Commission seek alternatives to mitigate the delay or supplement the size of the group working or seek further and fuller explanation for the delay?

ANSWER

Yes. The steps taken to mitigate the delay are described in the response to question 16.

QUESTION 18. What laws, regulations, and internal NRC procedures govern the opportunity for public comment on the Environmental Impact Statements for Early Site Permits?

ANSWER

Section 102 of the National Environmental Policy Act (NEPA) states, in part, that "[c]opies of such [environmental impact] statement and the comments and views of the appropriate Federal, State, and local agencies, which are authorized to develop and enforce environmental standards, shall be made available to the President, the Council on Environmental Quality and to the public..." The NRC implementing regulations for NEPA in 10 CFR 51.73 state, in part that "[e]ach draft environmental impact statement...will be accompanied by or include a request for comments on the proposed action and on the draft environmental impact statement...and will state where comments should be submitted and the date on which the comment period closes." NRC's procedures include opportunities for direct stakeholder interactions during the scoping process and during the public comment period on the draft EIS. A public meeting is encouraged as part of the scoping process, but there is no requirement that a public meeting be held during the public comment period. The draft EIS must also be provided to the EPA for comment.

QUESTION 19. What is the process for receiving, managing and responding to public comments?

ANSWER

In order to accommodate the fullest participation by the public, comments may be provided to the staff in a number of ways. After the staff issues a draft EIS, it holds a public meeting in the vicinity of the proposed site to describe the preliminary results of its review and to receive oral and written comments from members of the public. The meeting is transcribed so that all comments are documented. In addition, members of the public may submit comments in writing by mail, e-mail, or through a link on the NRC's web site.

Comments from the public meeting are extracted from the transcript of the public meeting, reviewed by the NRC staff, and grouped together based on subject matter. In a similar manner, the staff reviews comment letters and e-mails, extracts the comments, and groups them together. The staff then develops responses to the various groups of similar comments. The staff also makes changes to the draft EIS based on the comments, if appropriate (e.g., to correct factual errors or clarify the evaluation). The resulting document (the final EIS) is then published.

Questions Concerning Proposed Revision to Part 52

QUESTION 20. Does the Commission believe that Part 52 needs extensive revision? If so, on what experience or data does the Commission base this conclusion?

ANSWER

Revisions to Part 52 are currently being proposed, along with conforming changes throughout the NRC's regulations, to incorporate lessons learned from design certification and early site permit reviews in order to enhance the NRC's regulatory effectiveness and efficiency in implementing its new reactor licensing and approval processes. The proposed rule that the Commission approved for publication provides a reorganization of Part 52, implementing a uniform format and content for each of the subparts in Part 52, using consistent wording and organization of sections in each of the subparts, and making conforming changes elsewhere in the Commission's regulations to reflect the licensing and approval processes in Part 52.

QUESTION 21. When did the Commission provide direction to the staff to extensively revise Part 52? What direction did the Commission provide?

ANSWER

Anticipating that there would likely be lessons learned during the first use of the Part 52 licensing processes, the NRC had planned to update 10 CFR Part 52 after first using the standard design certification process. The proposed rulemaking action began with the issuance of SECY-98-282, "Part 52 Rulemaking Plan," on December 4, 1998. The Commission issued a Staff Requirements Memorandum on January 14, 1999, approving the NRC staff's plan for revising 10 CFR Part 52 to update and correct the rule based on "lessons learned" from the previous design certification rulemaking efforts and discussions with industry representatives on combined license review issues. Subsequently, the NRC obtained considerable stakeholder comment on its planned action, conducted three public meetings on the proposed rulemaking, and twice posted draft rule language on the NRC's rulemaking Web site before issuance of the initial proposed rule in July 2003.

Following the close of the public comment period on the July 2003 proposed rule, a number of factors led the NRC staff to question whether that proposed rule would meet the NRC's objective of improving the effectiveness of its processes for licensing future nuclear power plants. First, public comments identified several concerns about whether the proposed rule adequately addressed the relationship between Part 50 and Part 52, and whether it clearly specified the applicable regulatory requirements for each of the licensing and approval processes in Part 52. In addition, as a result of the NRC staff's review of the first three Early Site Permit applications, the staff gained additional insights into the early site permit process. The NRC also had the benefit of public meetings with external stakeholders on NRC staff guidance for the Early Site Permit and Combined License processes. As a result, the NRC staff decided that in addition to changes to address the foregoing, a reorganization of Part 52, implementing a uniform format and content for each of the subparts, and expansion of the original proposed rulemaking was desirable so that the agency could more effectively and efficiently implement the licensing and approval processes for future nuclear power plants under Part 52. The staff informed the Commission of its recommendation to expand the scope of the rulemaking when it requested an extension of the due date for the rulemaking in the Fall of 2004.

The Commission recently directed the NRC staff to withdraw the July 2003 proposed rule and publish the revised proposed rule for public comment. This revised proposed rule contains revisions to Part 52, as well as changes throughout the NRC's regulations, to ensure that all licensing and approval processes in Part 52 are addressed and to clarify the applicability of various requirements to each of the processes in Part 52. The staff plans to provide a final rule no later than October 2006 for Commission approval, as directed in SRM-SECY-05-0203.

25 Most Recently Licensed Plants

Plant	CP Application Filed	CP Issued	OL Application Filed	OL Issued	Elapsed Time (years)
Watts Bar 1	05/14/1971	01/23/1973	10/04/1976	02/07/1996	24.7
Comanche Peak 2	07/20/1973	12/19/1974	02/27/1978	04/06/1993	19.7
Comanche Peak 1	07/20/1973	12/19/1974	02/27/1978	04/17/1990	16.7
Seabrook 1	06/15/1973	07/07/1976	07/07/1976	03/15/1990	16.8
Limerick 2	02/26/1970	06/19/1974	02/26/1970	08/25/1989	19.5
Vogtle 2	08/01/1972	06/28/1974	08/01/1972	03/31/1989	16.7
South Texas 2	05/19/1974	12/22/1975	05/12/1978	03/28/1989	14.9
Braidwood 2	03/05/1973	12/31/1975	03/05/1973	05/20/1988	15.2
South Texas 1	05/19/1974	12/22/1975	05/12/1978	03/22/1988	13.8
Palo Verde 3	10/07/1974	05/25/1976	10/01/1979	11/25/1987	13.1
Beaver Valley 2	10/20/1972	05/03/1974	01/26/1983	08/14/1987	14.8
Braidwood 1	03/05/1973	12/31/1975	03/05/1973	07/02/1987	14.3
Nine Mile Point 2	03/08/1972	06/24/1974	01/31/1983	07/02/1987	15.3
Clinton	10/30/1973	02/24/1976	10/30/1973	04/17/1987	13.5
Vogtle 1	08/01/1972	06/28/1974	08/01/1972	03/16/1987	14.6
Byron 2	03/05/1973	12/31/1975	03/05/1973	01/30/1987	13.9
Shearon Harris 1	09/07/1971	01/27/1978	09/07/1971	01/12/1987	15.3
Perry 1	06/23/1973	05/03/1977	06/20/1980	11/13/1986	13.4
Hope Creek 1	02/27/1970	11/04/1974	02/27/1970	07/25/1986	16.4
Catawba 2	07/24/1972	08/07/1975	03/31/1981	05/15/1986	13.8
Palo Verde 2	10/07/1974	05/25/1976	10/01/1979	04/24/1986	11.5
Millstone 3	02/10/1973	08/09/1974	10/29/1982	01/31/1986	13.0
River Bend 1	07/08/1973	03/25/1977	04/24/1981	11/20/1985	12.4
Diablo Canyon 2	06/28/1968	12/09/1970	07/10/1973	08/26/1985	17.2
Limerick 1	02/26/1970	06/19/1974	02/26/1970	08/08/1985	15.5

ESBWR Design Certification Application Acceptance Review Checklist

Technical Information

The application for a design certification contains the following technical information required by 10 CFR Part 52.47:

		Yes	No
I.	The application contains the technical information which is required of applicants for construction permits and operating licenses by 10 CFR Part 20, Part 50 and its appendices, and Parts 73 and 100, which is technically relevant to the design and not site-specific [10 CFR 52.47(a)(1)(I)]. (see Attachment 1)	☐	☐
II.	The application contains a demonstration of compliance with any technically relevant portions of the Three Mile Island requirements set forth in 10 CFR 50.34(f) except paragraphs (f)(1)(xii), (f)(2)(ix) and (f)(3)(v) [10 CFR 52.47(a)(1)(ii)].	☐	☐
III.	The application contains the site parameters postulated for the design, and an analysis and evaluation of the design in terms of such parameters [10 CFR 52.47(a)(1)(iii)].	☐	☐
IV.	The application contains proposed technical resolutions of those Unresolved Safety Issues and medium- and high-priority Generic Safety Issues which are identified in the version of NUREG-0933 current on the date six months prior to application [NUREG-0933 Supplement 28, published August 2004] which are technically relevant to the design [10 CFR 52.47(a)(1)(iv)].	☐	☐
V.	The application contains a design-specific probabilistic risk assessment [10 CFR 52.47(a)(1)(v)].	☐	☐
VI.	The application contains proposed inspections, tests, analyses, and acceptance criteria which are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant which references the design is built and will operate in accordance with the design certification [10 CFR 52.47(a)(1)(vi)].	☐	☐
VII.	The application contains the interface requirements to be met by those portions of the plant for which the application does not seek certification. These requirements must be sufficiently detailed to allow completion of the final safety analysis and design-specific probabilistic risk assessment required by paragraph V above [10 CFR 52.47(a)(1)(vii)].	☐	☐
VIII.	The application contains justification that compliance with the interface requirements of paragraph VII above is verifiable through inspection, testing (either in the plant or elsewhere), or analysis. The method to be used for verification of interface requirements must be included as part of the proposed inspections, tests, analyses, and acceptance criteria required by paragraph VI above [10 CFR 52.47(a)(1)(viii)].	☐	☐

		Yes	No
IX.	The application contains a representative conceptual design for those portions of the plant for which the application does not seek certification, to aid the staff in its review of the final safety analysis and probabilistic risk assessment required by paragraph V above, and to permit assessment of the adequacy of the interface requirements called for by paragraph VII above [10 CFR 52.47(a)(1)(ix)].	☐	☐
X.	The application contains a level of design information sufficient to enable the Commission to judge the applicants' proposed means of assuring that construction conforms to the design and to reach a final conclusion on all safety questions associated with the design before the certification is granted. The information submitted includes performance requirements and design information sufficiently detailed to permit the preparation of acceptance and inspection requirements by the NRC, and procurement specifications and construction and installation specifications by an applicant. The Commission will require, prior to design certification, that information normally contained in certain procurement specifications and construction and installation specifications be completed and available for audit if such information is necessary for the Commission to make its safety determination [10 CFR 52.47(a)(2)].	☐	☐
XI.	The application contains any information beyond that required by 10 CFR 52.47 which the staff advised the applicant to submit with the design certification application [10 CFR 52.47(a)(3)]. This includes addressing issues discussed in SECY papers and SRMs needed to support the review of the application. (see Attachment 2)	☐	☐
XII.	The application contains an essentially complete nuclear power plant design except for site-specific elements such as the service water intake structure and the ultimate heat sink [10 CFR 52.47(b)(1) and 10 CFR 52.47(b)(2)(i)(A)(4)]; or there has been acceptable testing of an appropriately sited, full size, prototype [10 CFR 52.47(b)(2)(i)(B)].	☐	☐
XIII.	The performance of each safety feature of the design has been demonstrated through either analysis, appropriate test programs, experience, or a combination thereof [10 CFR 52.47(b)(2)(i)(A)(1)].	☐	☐
XIV.	Interdependent effects among the safety features of the design have been found acceptable by analysis, appropriate test programs, experience, or a combination thereof [10 CFR 52.47(b)(2)(i)(A)(2)].	☐	☐
XV.	Sufficient data exists on the safety features of the design to assess the analytical tools used for safety analyses over a sufficient range of normal operating conditions, transient conditions, and specified accident sequences, including equilibrium core conditions [10 CFR 52.47(b)(2)(i)(A)(3)].	☐	☐
XVI.	The application proposes the specific testing necessary to support certification of the design [10 CFR 52.47(b)(2)(ii)].	☐	☐

Procedural Requirements

		Yes	No
The design certification application meets the following procedural requirements:			
A.	The application follows the relevant sections of 10 CFR 50.4 [10 CFR 52.45(d)].		
	a. The application is addressed to the Document Control Desk and sent by mail, hand delivered, or sent by electronic submission in accordance with the requirements of 10 CFR 50.4(a).	◻	◻
	b. If the application is on paper, the submission must be the signed original ` [10 CFR 50.4(b)].	◻	◻
	c. The form of the application meets the requirements of 10 CFR 50.4(c).	◻	◻
B.	The application is submitted under oath or affirmation [10 CFR 50.30(b), 10 CFR 52.45(d)].	◻	◻
C.	The application for design certification must include an application for a final design approval [10 CFR 52.45(c)].	◻	◻
D.	The application includes an agreement limiting access to Classified Information [10 CFR 50.37].	◻	◻
E.	The application meets the provisions of 10 CFR 2 related to public availability including the provisions of 10 CFR 2.390 concerning proprietary information [10 CFR 50.39].	◻	◻

Technical information included in ESBWR design certification application:

PART 20--STANDARDS FOR PROTECTION AGAINST RADIATION

	Yes	No
20.1406 Minimization of contamination.		

PART 50--DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

		Yes	No
50.12	Specific exemptions.		
50.34	Contents of applications; technical information.		
50.34(a)	Preliminary safety analysis report.		
50.34(b)	Final safety analysis report.		
50.34(c)	Physical security plan.		
50.34(d)	Safeguards contingency plan.		
50.34(e)	Protection against unauthorized disclosure.		
50.34(f)	Additional TMI-related requirements.		
50.34(g)	Combustible gas control.		
50.34(h)	Conformance with the Standard Review Plan (SRP).		
50.34a	Design objectives for equipment to control releases of radioactive material in effluents--nuclear power reactors.		
50.36	Technical specifications.		
50.36a	Technical specifications on effluents from nuclear power reactors.		
50.44	Combustible gas control for nuclear power reactors.		
50.46	Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors.		
50.46a	Acceptance criteria for reactor coolant system venting systems.		
50.47	Emergency plans.		
50.49	Environmental qualification of electric equipment important to safety for nuclear power plants.		
50.55a	Codes and standards.		

		Yes	No
50.61	Fracture toughness requirements for protection against pressurized thermal shock events.		
50.62	Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants.		
50.63	Loss of all alternating current power.		
50.69	Risk-informed categorization and treatment of structures, systems and components for nuclear power reactors.		
Appendix A to Part 50--General Design Criteria for Nuclear Power Plants			
I. Overall Requirements:			
1 Quality Standards and Records			
2 Design Bases for Protection Against Natural Phenomena			
3 Fire Protection			
4 Environmental and Dynamic Effects Design Bases			
5 Sharing of Structures, Systems, and Components			
II. Protection by Multiple Fission Product Barriers:			
10 Reactor Design			
11 Reactor inherent Protection			
12 Suppression of Reactor Power Oscillations			
13 Instrumentation and Control			
14 Reactor Coolant Pressure Boundary			
15 Reactor Coolant System Design			
16 Containment Design			
17 Electric Power Systems			
18 Inspection and Testing of Electric Power Systems			
19 Control Room			
III. Protection and Reactivity Control Systems:			
20 Protection System Functions			
21 Protection System Reliability and Testability			
22 Protection System Independence			

	Yes	No
23 Protection System Failure Modes		
24 Separation of Protection and Control Systems		
25 Protection System Requirements for Reactivity Control Malfunctions		
26 Reactivity Control System Redundancy and Capability		
27 Combined Reactivity Control Systems Capability		
28 Reactivity Limits		
29 Protection Against Anticipated Operational Occurrences		
IV. Fluid Systems:		
30 Quality of Reactor Coolant Pressure Boundary		
31 Fracture Prevention of Reactor Coolant Pressure Boundary		
32 Inspection of Reactor Coolant Pressure Boundary		
33 Reactor Coolant Makeup		
34 Residual Heat Removal		
35 Emergency Core Cooling		
36 Inspection of Emergency Core Cooling System		
37 Testing of Emergency Core Cooling System		
38 Containment Heat Removal		
39 Inspection of Containment Heat Removal System		
40 Testing of Containment Heat Removal System		
41 Containment Atmosphere Cleanup		
42 Inspection of Containment Atmosphere Cleanup Systems		
43 Testing of Containment Atmosphere Cleanup Systems		
44 Cooling Water		
45 Inspection of Cooling Water System		
46 Testing of Cooling Water System		
V. Reactor Containment:		
50 Containment Design Basis		
51 Fracture Prevention of Containment Pressure Boundary		
52 Capability for Containment Leakage Rate Testing		

	Yes	No
53 Provisions for Containment Testing and Inspection		
54 Systems Penetrating Containment		
55 Reactor Coolant Pressure Boundary Penetrating Containment		
56 Primary Containment Isolation		
57 Closed Systems Isolation Valves		
VI. Fuel and Radioactivity Control:		
60 Control of Releases of Radioactive Materials to the Environment		
61 Fuel Storage and Handling and Radioactivity Control		
62 Prevention of Criticality in Fuel Storage and Handling		
63 Monitoring Fuel and Waste Storage		
64 Monitoring Radioactivity Releases		
Appendix B to Part 50--Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants		
Appendix E to Part 50--Emergency Planning and Preparedness for Production and Utilization Facilities		
Appendix G to Part 50--Fracture Toughness Requirements		
Appendix H to Part 50--Reactor Vessel Material Surveillance Program Requirements		
Appendix I to Part 50--Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion "As Low as is Reasonably Achievable" for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents		
Appendix K to Part 50--ECCS Evaluation Models		
Appendix S to Part 50--Earthquake Engineering Criteria for Nuclear Power Plants		

PART 73--PHYSICAL PROTECTION OF PLANTS AND MATERIALS

	Yes	No
73.55 Requirements for physical protection of licensed activities in nuclear power reactors against radiological sabotage.		

Additional issues to be addressed in ESBWR design certification application:

I. SECY-90-016 Issues:

A. Use of a Physically Based Source Term
B. Anticipated Transient Without Scram
D. Station Blackout
E. Fire Protection
F. Intersystem Loss-of-Coolant Accident
G. Hydrogen Control
H. Core Debris Coolability
I. High-Pressure Core Melt Ejection
J. Containment Performance
K. Dedicated Containment Vent Penetration
L. Equipment Survivability
M. Elimination of Operating-Basis Earthquake
N. Inservice Testing of Pumps and Valves

II. Other Evolutionary and Passive Design Issues (SECY-93-087):

A. Industry Codes and Standards
B. Electrical Distribution
C. Seismic Hazard Curves and Design Parameters
D. Leak-Before-Break
E. Classification of Main Steamlines in Boiling Water Reactors
F. Tornado Design Basis
G. Containment Bypass
H. Containment Leak Rate Testing
I. Post-Accident Sampling System
J. Level of Detail
K. Prototyping
L. ITAAC
M. Reliability Assurance Program
N. Site-Specific Probabilistic Risk Assessments and Analysis of External Events
O. Severe Accident Mitigation Design Alternatives
P. Generic Rulemaking Related to Design Certification
Q. Defense Against Common-Mode Failures in Digital Instrumentation and Control Systems
S. PRA Beyond Design Certification
T. Control Room Annunciator (Alarm) Reliability

III. Issues Limited to Passive Designs (SECY-93-087):
A. Regulatory Treatment of Nonsafety Systems in Passive Designs
B. Definition of Passive Failure
C. SBWR Stability
D. Safe Shutdown Requirements
E. Control Room Habitability
F. Radionuclide Attenuation
G. Simplification of Offsite Emergency Planning
H. Role of the Passive Plant Control Room Operator

IV. Confirmatory Items from NRC safety evaluation report for General Electric topical report NEDC-33083P regarding the application of TRACG Code to ESBWR LOCA analyses:

The applicant is to address the confirmatory items contained in the NRC safety evaluation report contained in NRC letter dated October 28, 2004, ADAMS Accession Number ML043000285.

V. Incorporate Operating Experience Into Design:

Operating experience is to be addressed in the design as requested by SRMs dated July 31, 1989, February 15, 1991, and March 5, 1991. This includes the operating experience discussed in NRC Bulletins (BLs) and Generic Letters (GLs). At a minimum, an ESBWR design certification application should address the BLs and GLs listed below:

<u>Bulletins</u>:

BL 79-02r2*	Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts
BL 79-08	Events Relevant to Boiling Water Reactors Identified During Three Mile Island Incident
BL 80-01	Operability of ADS Valve Pneumatic Supply
BL 80-03	Loss of Charcoal from Standard Type II, 2 Inch, Tray Absorber Cells
BL 80-05	Vacuum Condition Resulting in Damage to Chemical Volume Control System (CVCS) Holdup Tanks
BL 80-06	Engineered Safety Feature (ESF) Reset Controls
BL 80-08	Examination of Containment Liner Penetration Welds
BL 80-10	Contamination of Nonradioactive System and Resulting Potential for Unmonitored, Uncontrolled Release of Radioactivity to Environment
BL 80-12	Decay Heat Removal System Operability
BL 80-13	Cracking in Core Spray Spargers
BL 80-15	Possible Loss of Emergency Notification System with Loss of Offsite Power
BL 80-20	Failures of Westinghouse Type W-2 Spring Return To Neutral Control Switches
BL 80-21	Valve Yokes Supplied by Malcolm Foundry Company
BL 80-22	Automatic Industries, Model 200-500-008 Sealed Source Connectors
BL 80-24	Prevention of Damage Due To Water Leakage Inside Containment

BL 80-25	Operating Problems With Target Rock Safety-Relief Valves at BWRs
* r = Revision	
BL 81-01	Surveillance of Mechanical Snubbers
BL 81-02	Failure of Gate Type Valves to Close Against Differential Pressure
BL 81-02s1*	Failure of Gate Type Valves to Close Against Differential Pressure
BL 81-03	Flow Blockage of Cooling Water to Safety System Components by Corbicula Sp. (Asiatic Clam) and Mytilus Sp. (Mussel)
BL 82-04	Deficiencies in Primary Containment Electrical Penetration Assemblies
BL 83-06	Nonconforming Materials Supplied by Tube-Line Corporation Facilities
BL 84-01	Cracks in Boiling Water Reactor Mark 1 Containment Vent Headers
BL 84-03	Refueling Cavity Water Seal
BL 85-03	Motor-Operated Valve Common Mode Failure During Plant Transients Due to Improper Switch Settings
BL 85-03s1	Motor-Operated Valve Common Mode Failure During Plant Transients Due to Improper Switch Settings
BL 86-01	Minimum Flow Logic Problems That Could Disable RHR Pumps
BL 86-03	Potential Failure of Multiple ECCS Pumps Due to Single Failure of Air-Operated Valve in Minimum Flow Recirculation Line
BL 87-01	Thinning of Pipe Walls in Nuclear Power Plants
BL 87-02	Fastener Testing to Determine Conformance with Applicable Material Specifications
BL 87-02s1	Fastener Testing to Determine Conformance with Applicable Material Specifications
BL 87-02s2	Fastener Testing to Determine Conformance with Applicable Material Specifications
BL 88-04	Potential Safety-Related Pump Loss
BL 88-07	Power Oscillations in Boiling Water Reactors
BL 88-07s1	Power Oscillations in Boiling Water Reactors
BL 90-01	Loss of Fill-Oil in Transmitters Manufactured by Rosemount

BL 90-01s1 Loss of Fill-Oil in Transmitters Manufactured by Rosemount

* s = Supplement

BL 90-02 Loss of Thermal Margin Caused by Channel Box Bow

BL 91-01 Reporting Loss of Criticality Safety Controls

BL 91-01s1 Reporting Loss of Criticality Safety Controls

BL 92-01 Failure of Thermo-Lag 330 Fire Barrier System to Maintain Cabling in Wide Cable Trays and Small Conduits Free from Fire Damage

BL 92-01s1 Failure of Thermo-Lag 330 Fire Barrier System to Perform its Specified Fire Endurance Function

BL 93-02 Debris Plugging of Emergency Core Cooling Suction Strainers

BL 93-02s1 Debris Plugging of Emergency Core Cooling Suction Strainers

BL 93-03 Resolution of Issues Related to Reactor Vessel Water Level Instrumentation in BWRs

BL 94-01 Potential Fuel Pool Draindown Caused by Inadequate Maintenance Practices at Dresden Unit 1

BL 95-02 Unexpected Clogging of a Residual Heat Removal Pump Strainer While Operating in Suppression Pool Cooling Mode

BL 96-02 Movement of Heavy Loads Over Spent Fuel, Over Fuel in the Reactor Core, or Over Safety-Related Equipment

BL 96-03 Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling-Water Reactors

BL 2005-02 Emergency Preparedness and Response Actions for Security-Based Events

Generic Letters:

GL 80-34 Clarification of NRC Requirements for Emergency Response Facilities at Each Site

GL 80-113 Control of Heavy Loads

GL 81-03 Implementation of NUREG-0313, Rev. 1, Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping

GL 81-04	Emergency Procedures and Training for Station Blackout Events
GL 81-07	Control of Heavy Loads
GL 81-10	Post-TMI Requirements for the Emergency Operations Facility
GL 81-11	Comments on NUREG-0619
GL 81-20	Safety Concerns Associated with Pipe Breaks in the BWR Scram System
GL 81-37	ODYN Code Reanalysis Requirements
GL 81-38	Storage of Low-Level Radioactive Wastes at Power Reactor Sites
GL 82-09	Environmental Qualification of Safety-Related Electrical Equipment
GL 82-21	Technical Specifications for Fire Protection Audits
GL 82-23	Inconsistency Between Requirements of 10 CFR 73.40(d) and Standard Technical Specifications for Performing Audits of Safeguards Contingency Plans (Security Plan)
GL 82-27	Transmittal of NUREG-0763, "Guidelines for Confirmatory In-Plant Tests of Safety-elief Valve Discharges for BWR Plants," and NUREG-0783, "Suppression Pool Temperature Limits for BWR Containments"
GL 82-33	Supplement 1 to NUREG-0737 - Requirements for Emergency Response Capability
GL 82-39	Problems with the Submittals of 10 CFR 73.21 Safeguards Information for Licensing Review
GL 83-05	Safety Evaluation of "Emergency Procedure Guidelines, Revision 2," NEDO-24934, June 1982
GL 83-07	The Nuclear Waste Policy Act of 1982
GL 83-13	Clarification of Surveillance Requirements for HEPA Filters and Charcoal Adsorber Units in Standard Technical Specifications on ESF Cleanup Systems
GL 83-28	Required Actions Based on Generic Implications of Salem ATWS Events
GL 83-33	NRC Positions on Certain Requirements of Appendix R to 10 CFR 50
GL 84-15	Proposed Staff Actions to Improve and Maintain Diesel Generator Reliability
GL 84-23	Reactor Vessel Water Level Instrumentation in BWRs
GL 86-10	Implementation of Fire Protection Requirements

GL 87-06	Periodic Verification of Leak Tight Integrity of Pressure Isolation Valves
GL 87-09	Sections 3.0 and 4.0 of the Standard Technical Specifications (STS) on the Applicability of Limiting Conditions for Operation and Surveillance Requirements
GL 88-01	NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping
GL 88-14	Instrument Air Supply System Problems Affecting Safety-Related Equipment
GL 88-15	Electric Power Systems - Inadequate Control Over Design Processes
GL 88-16	Removal of Cycle-Specific Parameter Limits from Technical Specifications
GL 88-18	Plant Record Storage on Optical Disks
GL 88-20	Individual Plant Examination for Severe Accident Vulnerabilities - 10 CFR 50.54(f)
GL 88-20s1	Initiation of the Individual Plant Examination for Severe Accident Vulnerabilities - 10 CFR 50.54(f)
GL 88-20s2	Accident Management Strategies for Consideration in the Individual Plant Examination Process
GL 88-20s3	Completion of Containment Performance Improvement Program and Forwarding of Insights for Use in the Individual Plant Examination for Severe Accident Vulnerabilities
GL 88-20s4	Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities - 10 CFR 50.54(f)
GL 88-20s5	Individual Plant Examination of External Events for Severe Accident Vulnerabilities
GL 89-01	Implementation of Programmatic Controls for Radiological Effluent Technical Specifications in the Administrative Controls Section of the Technical Specifications and the Relocation of Procedural details of RETS to the Offsite Dose Calculation Manual or to the Process Control Program
GL 89-02	Actions to Improve the Detection of Counterfeit and Fraudulently Marketed Products
GL 89-04	Guidance on Developing Acceptable Inservice Testing Programs
GL 89-04s1	Guidance on Developing Acceptable Inservice Testing Programs
GL 89-06	Task Action Plan Item I.D.2 - Safety Parameter Display System - 10 CFR 50.54(f)
GL 89-07	Power Reactor Safeguards Contingency Planning for Surface Vehicle Bombs

GL 89-07s1	Power Reactor Safeguards Contingency Planning for Surface Vehicle Bombs
GL 89-08	Erosion/Corrosion-Induced Pipe Wall Thinning
GL 89-10	Safety-Related Motor-Operated Valve Testing and Surveillance
GL 89-10s1	Results of the Public Workshops
GL 89-10s3	Consideration of the Results of NRC Sponsored Tests of Motor-Operated Valves
GL 89-10s4	Consideration of Valve Mispositioning in Boiling Water Reactors
GL 89-10s5	Inaccuracy of Motor-Operated Valve Diagnostic Equipment
GL 89-10s6	Information on Schedule and Grouping, and Staff Responses to Additional Public Questions
GL 89-13	Service Water System Problems Affecting Safety-Related Equipment
GL 89-13s1	Service Water System Problems Affecting Safety-Related Equipment
GL 89-14	Line Item Improvements in Technical Specifications - Removal of the 3.25 Limit on Extending Surveillance Intervals
GL 89-15	Emergency Response Data System
GL 89-16	Installation of a Hardened Wetwell Vent
GL 89-18	Resolution of Unresolved Safety Issue A-17, “Systems Interactions in Nuclear Power Plants”
GL 89-19	Request for Action Related to Resolution of Unresolved Safety Issue A-47, “Safety Implication of Control Systems in LWR Nuclear Power Plants,” Pursuant to 10 CFR 50.54(f)
GL 89-22	Potential for Increased Roof Loads and Plant Area Flood Runoff Depth at Licensed Nuclear Power Plants Due to Recent Change in Probable Maximum Precipitation Criteria Developed By the National Weather Service
GL 90-09	Alternative Requirements for Snubber Visual Inspection Intervals and Corrective Actions
GL 91-03	Reporting of Safeguards Events
GL 91-04	Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle
GL 91-05	Licensee Commercial-Grade Procurement and Dedication Programs

GL 91-06	Resolution of Generic Issue A-30, “Adequacy of Safety-Related DC Power Supplies,” Pursuant to 10 CFR 50.54(f)
GL 91-10	Explosives Searches at Protected Area Portals
GL 91-11	Resolution of Generic Issues 48, “LCOs for Class 1E Vital Instrument Buses,” and 49, ”Interlocks and LCOs for Class 1E Tie Breakers” Pursuant to 10 CFR 50.54(f)
GL 91-14	Emergency Telecommunications
GL 91-16	Licensed Operators’ and Other Nuclear Facility Personnel Fitness for Duty
GL 91-17	Generic Safety Issue 29, “Bolting Degradation or Failure in Nuclear Power Plants”
GL 92-01r1	Reactor Vessel Structural Integrity
GL 92-04	Resolution of the Issues Related to Reactor Vessel Water Level Instrumentation in BWRs Pursuant to 10 CFR 50.54(f)
GL 92-08	Thermo-Lag 330-1 Fire Barriers
GL 93-06	Research Results on Generic Safety Issue 106, "Piping and the Use of Highly Combustible Gases in Vital Areas"
GL 94-02	Long-Term Solutions and Upgrade of Interim Operating Recommendations for Thermal-Hydraulic Instabilities in Boiling Water Reactors
GL 94-03 I	Intergranular Stress Corrosion Cracking of Core Shrouds in Boiling Water Reactors
GL 95-07	Pressure Locking and Thermal Binding of Safety- Related Power-Operated Gate Valves
GL 96-01	Testing of Safety-Related Logic Circuits
GL 96-04	Boraflex Degradation in Spent Fuel Pool Storage Racks
GL 96-05	Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves
GL 96-06	Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions
GL 96-06s1	Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions

GL 97-04	Assurance of Sufficient Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps
GL 98-04	Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System After a Loss-of-Coolant Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment
GL 99-02	Laboratory Testing of Nuclear- Grade Activated Charcoal
GL 2003-01	Control Room Habitability

Representative List of Federal, State, and Local Authorizations and Consultations (North Anna ESP example)

Agency	Authority	Requirement	Activity Covered
Federal Aviation Administration	49 USC 1501 14 CFR 77.13	Construction Notice	Notice of erection of structures (>200 feet) potentially impacting air navigation
NRC	Atomic Energy Act, 10 CFR 51, 10 CFR 52.17	EIS	Environmental effect of construction and operation of a reactor
NRC	10 CFR Part 52, Subpart C	Combined License	NRC requirements and procedures applicable to issuance of combined licenses for nuclear power facilities
NRC	10 CFR Part 52, Subpart A	Early Site Permit	NRC requirements and procedures applicable to issuance of Early Site Permits for approval of a site for one or more nuclear power facilities
NRC	10 CFR Part 30	Byproduct License	NRC license to possess byproduct materials
NRC	10 CFR Part 40	Source Material	NRC license to possess source material
NRC	10 CFR Part 70	License	NRC license to possess special nuclear material and nuclear fuel
SCC			Approval of the purchase or lease of the site
Army Corps of Engineers (ACE)	Clean Water Act (CWA) 33 USC 1251	Section 404 Permit	Disturbing or crossing wetland areas or navigable waters
ACE	Rivers and Harbors Act 33 USC 403	Section 10 Permit	Impacts to navigable waters of the United States

Agency	Authority	Requirement	Activity Covered
Fish and Wildlife Service (FWS) and National Oceanographic and Atmospheric Administration (NOAA) Fisheries Service	Endangered Species Act 16 USC 1531	Consultation regarding potential to adversely impact protected species	Consultation concerning potential impacts to threatened and endangered species
FWS	Migratory Bird Treaty Act 16 USC 703	Consultation	Consultation concerning potential impacts to migratory birds
Virginia State Corporation Commission	Code of Virginia 56-580D	Permit	Approval for construction of new generating facility
Virginia Department of Environmental Quality (VDEQ)	9 VAC 5-20-160	Registration	Annual re-certification of air emission sources
VDEQ	Clean Air Act Title V 9 VAC 5-80-50	Operating Permit	Operation of air emission sources
VDEQ	9 VAC 5-80-120	Minor Source - General Permit	Construction and operation of minor air emission sources
VDEQ	CWA 9 VAC 25-10	Virginia Pollutant Discharge Elimination System Permit (VPDES)	Regulate limits of pollutants in liquid discharge to surface water
VDEQ	9 VAC 25-150	General Permit Registration Statement for storm water discharges from industrial activity (VAR5)	General permit to discharge storm water from site during operations
VDEQ	9 VAC 25-180	General Permit Notice of Termination (NOT) for storm water discharges from construction activities (VAR4)	Termination of coverage under the general permit for storm water discharge from construction site activities

Agency	Authority	Requirement	Activity Covered
VDEQ	9 VAC 25-180	General Permit NOT for storm water discharges from industrial activity (VAR5)	Termination of coverage under the general permit for storm water discharge associated with operational site activities
VDEQ	9 VAC 25-210	Virginia Water Protection Permit (Individual or General)	Permits to dredge, fill, discharge pollutants into or adjacent to surface water. Joint application with USACE Section 404 permit.
VDEQ	CWA	Section 401 Certification	Compliance with water quality standards
VDEQ	CWA 9 VAC 25-220	Surface Water Withdrawal Permit	Permit to draw water from Lake Anna (unless otherwise regulated by State Water Control Board)
VDEQ	Coastal Zone Management Act, Section 307	Consistency determination	Compliance with Virginia Coastal Program
VDEQ	Virginia Coastal Resources Management Program	Consistency determination	Compliance with Virginia Coastal Program
VDEQ	CWA 9 VAC 25-180	General Permit Registration Statement for storm water discharges from construction activities (VAR10)	General permit to discharge storm water from site during construction
Virginia Department of Historical Resources	National Historic Preservation Act 36 CFR 800	Cultural Resources Survey/Review	Confirm ESP site does not contain protected historic/cultural resources
Virginia Marine Resources Commission	9 VAC 25-210	Permit	Permit to fill submerged land. Joint application with ACE Section 404 permit.
Lake Anna Special Area Committee		Conditional Land Use Approval	Local land use approval - Lake Overlay District

February 10, 2006

Mr. David A. Christian
Senior Vice President and Chief Nuclear Officer
Dominion Resources Services
Innsbrook Technical Center
5000 Dominion Blvd.
Glen Allen, VA 23060-6711

SUBJECT: NORTH ANNA EARLY SITE PERMIT (ESP) APPLICATION REVIEW SCHEDULE (TAC NOS. MC1126 AND MC 1128)

Dear Mr. Christian:

On January 13, 2006, Dominion Nuclear North Anna, LLC (Dominion) submitted a supplement to its application for an early site permit (ESP) for the North Anna ESP site. The supplement proposes to change the cooling system for proposed Unit 3 and increase the power level for both proposed Units 3 and 4 from 4300 MWt to 4500 MWt. The purpose of this letter is to inform you of the results of the U.S. Nuclear Regulatory Commission (NRC) staff's initial review of the supplement to the ESP application and provide a schedule for the completion of the review. Dominion's decision to revise the application has resulted in a revision to the review schedule.

The cooling system change for proposed Unit 3 from once through cooling to a closed cooling system and the power level increase for both proposed Units 3 and 4 from 4300 MWt to 4500 MWt are substantial changes. These substantial changes to your application require the NRC to issue, in accordance with 10 CFR 51.72, a supplement to its draft environmental impact statement (EIS). The changes also warrant a supplement to the final safety evaluation report (SER). These supplements will focus on the impacts of the above changes. After analyzing the changes in your submittal, the staff will issue its supplement to the draft EIS for public comment and issue its supplemental final SER. After the comments are received and analyzed, the final EIS will be issued as shown in the schedule in Attachment 2.

The NRC staff has reviewed the contents of your supplement and has identified several key areas in which the staff needs additional information to complete its review. 10 CFR 51.45 requires that the environmental report contain a description of the proposed action and its impact on the environment. In the supplement, Dominion failed to adequately describe the operation of the new cooling system, including its interactions with the environment and its effects on reactor site criteria. Dominion also failed to adequately address the impact of the new cooling system and associated consumptive water use on downstream users and aquatic biota downstream of the dam. These issues and the other key areas for which the staff needs additional information are discussed in detail in Attachment 1 to this letter. In a separate letter, the staff will request information on additional issues identified during the review.

We request that you provide a complete revised application addressing the staff's information needs identified in Attachment 1. Once you have adequately addressed the issues, the staff expects to issue the supplemental final SER and the final EIS for the North Anna ESP application within 260 days in accordance with the schedule in Attachment 2 to this letter. In order for the schedule in Attachment 2 to be met, Dominion must be able to meet its milestones and provide a high quality revised application.

In your letters dated October 24 and November 22, 2005, you stated that you have made these design changes partly due to the concerns raised by state regulatory bodies. Please confirm that the concerns raised by the state agencies have been resolved, and that another substantial design change, which would impact both the NRC's and Dominion's schedules and resources, will not be necessary.

The staff plans to arrange a meeting with you within two weeks of the issuance of this letter to discuss in detail the key issues identified in Attachment 1 and the additional information requests that will be made under separate cover. If you have any questions on this matter, please contact the NRC Project Manager, Nitin Patel, at 301-415-3201 or nxp1@nrc.gov.

Sincerely,

/RA W. Beckner for:/

David B. Matthews, Director
Division of New Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No.: 52-008

Attachments :As stated

cc w/atts: See next page

Staff Information Needs Identified In the Review
of Dominion Nuclear North Anna, LLC
Supplement to Early Site Permit (ESP) Application
For the North Anna ESP Site

In reviewing an application for an ESP, the NRC staff makes certain decisions on the physical suitability of a specific site for the construction and operation of a nuclear power plant and the environmental impacts of plant construction and operation. The ESP application and review process makes it possible to evaluate and resolve safety and environmental issues related to siting before the applicant makes large commitments of resources. Before issuing an ESP, the NRC must first prepare both a safety evaluation report (SER) and an environmental impact statement (EIS). The purpose of the SER is to document the NRC staff's findings regarding site safety characteristics and emergency planning. The purpose of the EIS is to address questions regarding the impact of the proposed new reactor(s) on the environment. One of the siting issues for North Anna is water use and quality. In the previous draft EIS, the NRC evaluated the originally proposed once through cooling system for proposed Unit 3. The Commonwealth of Virginia expressed concerns about the consumptive water use and thermal impacts of the once through cooling system on Lake Anna and downstream users. In response, Dominion revised its application on January 13, 2006, to use a closed cycle cooling system consisting of wet and dry cooling towers for proposed Unit 3.

10 CFR 51.45 requires that the environmental report contain a description of the proposed action and its impact on the environment. In the supplement, Dominion failed to describe the operation of the new cooling system, including its interactions with the environment and its effects on reactor site criteria. Specifically, the description of when the dry cooling system for proposed Unit 3 will be used is too vague for the staff to determine the impacts on Lake Anna, downstream users or aquatic life downstream of the dam. In order to disclose the impacts, the staff will need to know the maximum amount of water proposed Unit 3 will use at the lake levels which correspond to changes in flow rates from the North Anna Dam.

In addition, deferral of analysis of environmental issues to the COL application cannot be done for issues that affect siting, such as cooling tower impacts. If design information is lacking, then the applicant should make reasonable assumptions about the design, so that the staff can evaluate the impacts. If the assumptions are bounding at the COL stage no further analysis will be required. If, the assumptions prove not to be bounding at the COL stage, then the staff will evaluate the new and potentially significant information.

The following are the key areas in which the staff needs additional information to complete its review**:**

1. Description of the Operation of the New Cooling System Design

The supplement includes a limited description of the operation of the closed-cycle dry and wet tower cooling system. Please provide a detailed description of the operation of the new cooling design, including interactions with the environment and effects on reactor site criteria.

Additional information for operational details identified to date include the following:

ER Section 5.2.2.4, "Proposed Practices to Minimize or Avoid Impacts"

This section does not provide sufficient detail to determine water usage and therefore the impact on the lake level and downstream users.

SSAR Section 2.4, "Hydrology"

This section does not provide sufficient information for the staff to assess the reliability of the hybrid cooling tower system insofar as its use affects the reliance of Unit 3 on its emergency cooling system.

2. Evaluation of Aquatic Impacts

The new cooling system design reduces the thermal impacts on the lake. However, there is insufficient evaluation on how the change to the cooling system and consumptive water use will affect aquatic biota and downstream users. In particular, the supplement should provide more detail on the impact of the new cooling system on striped bass in the lake and downstream of the dam.

3. Deferral of Required Analyses to the Combined License (COL) Application

The staff does not need to know the detailed reactor design at the ESP stage. However, the staff needs sufficient information to analyze and disclose the impacts to the environment. The staff must have sufficient information on the impacts to the environment of the proposed action to allow the staff to make a comparison to alternate sites. If design level information is lacking, the applicant should make reasonable assumptions about the potential design and evaluate the impact based on those assumptions. The following are the deferrals identified to date:

SSAR Chapter 15, "Accident Analysis and ER Section 7.1.3, "Source Terms"

In your letter dated November 22, 2005, you stated that you selected the ESBWR as your reactor design and that your supplement will fully address changes to the North Anna ESP application based on ESBWR design information provided in GE's design certification application.

The ABWR design is certified under 10 CFR Part 100, Subpart A and 10 CFR 50.34(a)(1)(i), while the ESBWR design is not certified and must be evaluated under 10 CFR Part 100, Subpart B and 10 CFR 50.34(a)(1)(ii). Please describe the design basis accidents, the reactor accident source terms, and the design-specific χ/Q values for the ESBWR design. Demonstrate that the reactor accident source term plant parameter envelope (PPE) values specified in the application are still appropriate, and that the radiological doses consequence at the proposed ESP site would meet the requirements of 10 CFR 50.34.

ER Section 5.3.3.1, “Heat Dissipation to the Atmosphere”

This section defers the analysis of fogging and salt deposition to the COL application. Please provide a detailed analysis, including reasonable assumptions for design features for mitigating the effects of fogging and salt deposition, so the staff can evaluate the impacts of fogging and salt deposition.

ER Section 5.8.1.2, “Noise”

This section concluded that the noise associated with the new cooling design would not cause adverse offsite impacts and that a noise study would be described in the COL application. Describe calculations and assumptions used to estimate noise levels at the exclusion area boundary (EAB) and closest residence. Include initial sound levels (background and cooling towers), the number of sources, distances, and attenuation factors considered in reaching a conclusion even if not included in the calculations.

4. Sections of Application Identified as Unaffected

The supplement provides no justification why the sections identified as unaffected by the change to the cooling system and the increase in power level are unaffected. For example, why is ER ection 7.2, Severe Accidents, not affected by the increase in power from 4300 - 4500 MWt. Provide justification for why sections identified as unaffected are not affected by the change in power level or the cooling system.

5. State Permits

In your letters dated October 24 and November 22, 2005, you stated that you have made these design changes partly due to the concerns raised by state regulatory bodies. Please confirm that the concerns raised by the state agencies have been resolved, and that another substantial design change, which would impact both NRC and Dominion schedules and resources, will not be necessary.

Milestone	Elapsed Time
Applicant submits Revision 6 of the early site permit application which addresses the key issues in Attachment 1	**T = 0**
Press Release announcing receipt and availability	T + 14 days
Safety requests for additional information (RAIs) issued to the applicant (optional)	T + 20 days
Federal Register Notice (FRN) published for Notice of Intent to prepare a supplement to the draft environmental impact statement (EIS)	T + 28 days
Applicant submits responses to safety RAIs	**T + 35 days**
Site Audit	T + 42 days
Applicant submits application final revision of the Site Safety Analysis Report	**T + 65 days**
Notice of Availability/Supplemental Draft EIS Issued, Start of Comment Period	T + 98 days
Supplemental Final safety evaluation report (SER) issued	T + 115 days
Public meeting to discuss draft EIS	
End of draft EIS comment period	T + 143 days
Advisory Committee on Reactor Safeguards (ACRS) Full Committee meeting on Supplemental Final SER	T + 145 days*
ACRS final letter to Executive Director for Operations	T + 170 days*
Supplemental Final SER issued as NUREG	T + 215 days*
Final EIS Issued	T + 260 days

* Milestones depend on ACRS availability (ACRS Full Committee does not meet in August or January)

It will take nominal 12 months for Atomic Safety and Licensing Board decision and Commission decision to issue Early Site Permit.