**From:** <john.hufnagel@exeloncorp.com>
**To:** <dja1@nrc.gov>, <rkm@nrc.gov>
**Date:** 03/09/2006 5:30:45 PM
**Subject:** Three additional Q&A Responses

Donnie/Roy,

Attached are three of the remaining 11 questions that were to be answered. Please confirm you have received and provide to the appropriate personnel. They are in the Q&A database. Thanks.

- John.

<<Q&As 358, 167 and 355.pdf>>

**Mail Envelope Properties** (4410AC84.9B1 : 22 : 18865)

**Subject:** Three additional Q&A Responses
**Creation Date:** 03/09/2006 5:29:52 PM
**From:** <john.hufnagel@exeloncorp.com>

**Created By:** john.hufnagel@exeloncorp.com

**Recipients**
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| MESSAGE | 1136 | 09 March, 2006 5:29:52 PM |
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| Q&As 358, 167 and 355.pdf | 71009 | |
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**Options**

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| **Expiration Date:** | None |
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| **Reply Requested:** | No |
| **Return Notification:** | None |
| **Concealed Subject:** | No |
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# *NRC Information Request Form*

***Item No***
AMP-358

***Date Received:*** 2/17/2006
***Source*** AMP Audit
***Status:*** Open

***Topic:***
CUF Reevaluation

***Document References:***
3.1

***NRC Representative*** Chang, Ken

***AmerGen (Took Issue):*** Warfel, Don

***<u>Question</u>***

QUESTIONS OF RORC MEETING (06-03) REPORT

As part of the review for AMP B.3.1, Metal Fatigue of Reactor Coolant pressure boundary the project team reviewed OC's PORC meeting (06-03) report, summarized the presentation, and reviewed OC-2006 E-001, Rev O, Revised Method for Determination of Fatigue Cumulative Usage Factor. OC used modern codes and revised STET the acceptance criteria for fatigue CUF. The PORC disposition is approved with recommendations with conditions. The project team does not question the use of the modern code, since it is a reasonable step to take, but has the following questions requiring clarification or justification.:

1.) Some RPV components are designed to a criterion established by GE specification 21A1105. Please provide a copy for NRC Staff review.

2.)The project team agrees that the design code of record does not require or specify fatigue analysis requirements. Nor were there any regulating design requirements for fatigue analysis at the time of design. An explanation is requested as to why GE included a prudent measure to limit the CUF to 0.8. Why didn't GE allow CUF of 1.0? Was CUF of 1.0-0.8=0.2 intentionally reserved for margin" The PORC report stated that this is not considered as a departure from the design (CUF 1.0) methodology. Please justify the statement.

3.) PORC question (2) states that : this activity involves a change to the methodology for the determination of the Fatigue CUF. What change does it refer to? As for determination of CUF=SUM (ni/ Ni) where ni is actual on design cycles and Ni is the allowable cycles for the I- th transient pair. Please clarify.

4) It seems to the project team that there is no change in methodology. The only thing changed is the CUF limit (from 0.8 to 1.0)) Was GE consulted to verify that it is acceptable w/o violating some original design concerns.

5) If OC changes the CUF from, 0.8 (design) to 1.0 for LR, how could one conclude that this activity has no adverse affect w/o justification? If they change from 0.8 to 0.7, the logic is obvious.

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6) Is GE SPEC 21A1105 voided? If so, what is the new spec OC used today for the PEO?

7) OC credited the new fatigue analysis as justification to the change of CUF. Please consider, if everything (condition) remains unchanged, if the original design meets CUF of 0.8, naturally, one will meet CUF if 1.0 today. What is the purpose of these analysis? Why don't you show that the CUF today is less than 0.8 but will be allowed to go up to 1.0 including environmental impact for the PEO?

8) The team would like to review the basis of justifying the CUF for FW Nozzle and Recir. Outlet Nozzle & RPV outlet.

***Assigned To:*** May, Mike

***Response:***

1. A copy of GE Specification 21A1105 was supplied to the NRC Staff during the Friday February 17, 2006 breakout session.

2. a) From UFSAR section 5.3.1.1, the following statement provides the basis for the General Electric method of performing fatigue analysis for the Oyster Creek reactor vessel; "For reactor pressure vessels designed and built prior to the adoption of the ASME Boiler and Pressure Vessel Code Section III, the General Electric Company developed a method for performing a fatigue analysis which would provide assurance that vessels installed in General Electric designed nuclear power plants would safely withstand all anticipated operating and transient conditions, both normal and emergency. This method was based upon the method of analysis developed for Naval reactors and upon industry's experience using it." The UFSAR also concludes that the General Electric Specification defined analysis results in a completed vessel for the Oyster Creek plant, which has safety margins that are generally equivalent to those which would result from using Section III methodology. General Electric's selection of a cumulative usage factor limit of 0.8 (versus 1.0) was to assure the Oyster Creek reactor pressure vessel design would remain bounded by the pending ASME Section III methodology and acceptance criterion. There is no evidence that consideration was given to reserving margin for any other reason (e.g., for system transients or unspecified cyclic conditions not considered in original analysis). The reanalyzed fatigue usage factors were performed to the ASME Section III requirements to demonstrate acceptability to the corresponding acceptance limit of 1.0.

b) The Exelon 50.59 evaluations reviewed if using ASME Section III instead of the methods by GE to calculate fatigue usage represented a departure from a method of evaluation described in the UFSAR used in establishing design bases. The OC procedure for preparing 50.59 evaluations, based on NEI 96-07, provides the guidance that: Use of a new NRC-approved methodology (e.g. ASME Section III) to reduce uncertainty, provide more precise results, or other reason is not a departure from a method of evaluation described in the UFSAR, provided such use is (a) based on sound engineering practice, (b) appropriate for the intended application, and (c) within the limitations of the applicable SER. Oyster Creek is using the ASME Boiler and Pressure Vessel Code Section III methodology to revise its design basis fatigue analyses for the reactor vessel; and the NRC has approved the use of ASME Boiler and Pressure Vessel Code Section III via 10CFR50.55a, which is within the limitations of the Oyster Creek Licensing Basis. Therefore, implementing the ASME Boiler and Pressure Vessel Code Section III method for analyzing fatigue is not considered a departure from a method of evaluation described in the UFSAR.

3. The licensing change allows Oyster Creek to revise design basis analysis from the methods described in GE specification 21A1105 to the NRC-approved methods of the ASME Boiler and Pressure Vessel Code Section III. The licensing basis change provides Oyster Creek the ability to implement revised analysis to establish new allowable cycles [N(i)], using the methods described in ASME Boiler and Pressure Vessel Code Section III. The difference in methodology is primarily associated with the difference between the s-N fatigue curve provided in the GE specification and the fatigue curve in the ASME Section III code. The process of summing transient pairs to determine total fatigue usage remains unchanged.

4. As part of the preparation of the Oyster Creek License Renewal application, limiting fatigue analyses of the reactor pressure vessel prepared per the original GE purchase specification for the RPV have been revised in accordance with the NRC approved ASME Boiler and Pressure Vessel Code Section III as permitted by Appendix L of ASME Section XI. As stated in Appendix L the new fatigue usage values are compared to 1.0. This is not only a change in acceptance limit but also a change in methodology, since fatigue usage factors were revised using the fatigue curve in ASME Section III instead of the fatigue curve provided in the GE specification.

Oyster Creek has assumed the responsibility of the RPV design basis analysis in accordance with the Code requirements, and therefore, GE concurrence of the changes is not required nor was it requested..

5. Oyster Creek has revised the fatigue analysis for the limiting RPV locations in accordance with the methods established in NRC approved ASME Boiler and Pressure Vessel Code Section III, as permitted by ASME Section XI IWB-3740. As stated in ASME XI Appendix L the revised usage factor are compared to 1.0. Since all of the revised usage factors are less than the acceptance limit there are no adverse effects.

6. The GE specification (21A1105) is still the current specification for the RPV. This specification will be updated to reflect the change in methodology as part the design change process.

7.As part of the effort for License Renewal the current licensing basis RPV fatigue analysis was evaluated to demonstrate satisfactory results for the period of extended operation. When the current licensing basis RPV fatigue analysis was reevaluated, using actual thermal cycles based on plant data, it was determined that for some locations the forty-year fatigue usage may exceed the 0.8 acceptance limit imposed by the GE spec. These locations required a more refined analysis. Under the rules of 10CFR50.55a and Section XI, Subsection IWB, the Licensee is allowed to use Appendix L of Section XI to analyze the effects of fatigue on components. Appendix L directs that ASME Section III fatigue usage factor evaluation procedures be used to determine if they are acceptable for continued service. The fatigue usage factors for the reanalyzed components are less than 0.8 before environmental effects are included for License Renewal. However, there is no technical basis not to compare the usage factors to 1.0 since Appendix L establishes 1.0 as the appropriate acceptance limit. Age.

8. The revised analysis for the above components can be found in Exelon Design Analysis SIA# OC-05Q-303 Revision 1. The appropriate fatigue analyses are available to the audit teams at the station.

***LRCR #:*** ***LRA A.5 Commitment #:***

***IR#:***

***Approvals:***

| | | |
|---|---|---|
| ***Prepared By:*** | May, Mike | 3/ 2/2006 |
| ***Reviewed By:*** | Beck, George | 3/ 2/2006 |
| ***Approved By:*** | Polaski, Fred | 3/ 2/2006 |

***NRC Acceptance (Date):***

# NRC Information Request Form

*Item No*
AMR-167

*Date Received:* 10/31/2005

*Source* AMR Audit

*Status:* Open

*Topic:*
Reactor Building Drywell Shield Wall

*Document References:*
3.5.2.2.2

*NRC Representative* Morante, Rich

*AmerGen (Took Issue):*

***Question***

More information is needed about the elevated temperature condition in the reactor building drywell shield wall. When was the condition first discovered? What was the extent of the elevated temperature region and what was the extent of the cracked region (distribution, length, width of cracks) when first discovered? When did NRC conclude that this condition is acceptable? Did this conclusion consider the remaining operating life of OC at that time? Describe the monitoring program, including the dates and quantitative results obtained, since NRC acceptance of the condition. Currently, what is the extent of the elevated temperature region and what is the extent of the cracked region (distribution, length, width of cracks)? Has there been a need to conduct re-analysis or make any repairs? Is the LR commitment under the OCGS SMP greater than, equal to, or less than the condition monitoring activities currently being conducted to satisfy the NRC staff's recommendation?

Follow-Up Question:

As follow-up to the applicant's response, the project team reviewed References 3 and 5, and ABB Impell Corporation Report # 03-0370-1341, Oyster Creek Nuclear Generating Station Structural Evaluation of the Spent Fuel Pool, Rev. 0, June 29, 1992. Based on its review, the project team has a concern that several potential aging issues may not have not been adequately addressed, in consideration of an additional 20 years of operation. These relate to known degradation of the drywell shield wall (DSW), the biological shield wall (BSW), and the spent fuel pool supporting structural elements.

The applicant is requested to review Ref. 3 and describe how it has implemented the following elements of the staff's SER:

(1) The staff's crack width acceptance criterion (0.02), above which repairs should be made to prevent water intrusion and potential corrosion of rebar in the drywell shield wall.

(2 The OCGS statement in Ref. 4 that it is developing procedures for monitoring the condition of the DSW during each refueling outage.

(3) The OCGS statement in Ref. 4 that it has assigned a structural-system engineer to the OCGS site

who is responsible for ensuring that the structures at the site are monitored and evaluated. (Ref. 3 states: The staff believes that this blanket commitment by the licensee, if properly implemented, would ensure the continued function of the BSW.)

The applicant is also requested to address the conclusion in the cited IMPELL Report, Section 5.4 Conclusions, (4), related to the effects of consolidated fuel loads.
Has OCGS implemented a fuel rack change that increases the total fuel load in the spent fuel pool?

The applicant is further requested to address the cracking in the spent fuel area that the cited analysis predicted and compared to actual observations of cracking. What is the applicant's aging management commitment for these cracks.

***Assigned To:*** Ouaou, Ahmed

***Response:***

More information is needed about the elevated temperature condition in the reactor building drywell shield wall.
a) When was the condition first discovered?
Response: The drywell shield wall elevated temperature concern surfaced in early to mid-1980's. The issue was evaluated as part of NUREG-0822, Integrated Plant Safety Assessment, Systematic Evaluation Program, Oyster Creek Nuclear Generating Station, January 1983, Topic III-7.B (Ref. 1)

b) What was the extent of the elevated temperature region and what was the extent of the cracked region (distribution, length, width of cracks) when first discovered?
Response: A review of the current licensing basis information did not identify documents that provide details on the extent of the cracked region when it was first discovered in mid-1980's. We were able to conclude that the condition of the wall was monitored after it was discovered. However no specific criteria such as distribution, width, and length of cracks was not identified. The earliest document that provides this information is an inspection report prepared in 1994. This report has been used since 1994 as a benchmark against which subsequent observed shield wall condition is evaluated. Observed cracks on the outside of drywell shield wall, as documented in the 1994 inspection report, show that the entire shield wall above elevation 95'-3" may be affected by the elevated temperature. Distribution of the cracks is generally random. Crack widths are generally hairline; with no cracks wider than 1/32".

c) When did NRC conclude that this condition is acceptable?
Response: The NRC Staff evaluation of information submitted by GPU, the previous owner of Oyster Creek, on the drywell shield wall elevated temperate began in 1986. In its Safety Evaluation dated October 24, 1986 (Ref. 2), the Staff required further investigation to complete its evaluation. GPU transmitted the requested information, in several correspondences between 1990 through 1993. The Staff completed its review of the submitted information and concluded in a Safety Evaluation dated May 11, 1994 that the drywell shield wall is capable of performing its intended function (Ref. 3).

d) Did this conclusion consider the remaining operating life of OC at that time?

Response: The May 11, 1994 Safety Evaluation (Ref. 3) did not specify that the conclusion is based on the remaining operating life of Oyster Creek. Such a conclusion would have been considered a TLAA and identified/evaluated in LRA Section 4.

e) Describe the monitoring program, including the dates and quantitative results obtained, since NRC acceptance of the condition.
Response: As recommended by the NRC Staff in May 11, 1994 Safety Evaluation, Oyster Creek implemented a periodic crack monitoring program. The program consists of visual inspection of drywell shield wall above elevation 95'-3" every refueling outage (Ref. 4). The benchmark inspection was conducted in April 1994 to record the surface condition of the drywell shield wall, including the crack patterns, crack length, and width.

On October 1996, during the refueling outage, a second inspection was performed to assess the condition of the drywell shield wall while the reactor cavity is flooded with water. No changes to the cracks or water stain were observed.

A similar inspection was performed during 1996 refueling outage. The following conditions were observed,
1. The crack pattern remained unchanged
2. Some observable hair line cracks may be newly developed or may be un-recorded from the last inspection
3. No cracks wider than 1/32" were observed
4. No spalling or new scaling/peeling was observed
5. There is evidence of growing water stains from the pipe penetration and cracks. Engineering evaluation concluded that this condition is local and has no impact on structural integrity of the wall.

The 1998 inspection concluded that
1. The crack pattern remained unchanged from previous inspection.
2. The inspector noted that the bottom 8' of the wall was repainted since last inspection and he could not observe new fine cracks . However, the inspector observed no new significant cracks.
3. Inspection of the wall after plant shutdown showed that the surface cracking were mostly closed and no excessive cracking
4. No concrete spalling was observed
5. New water stains were observed. However no significant leaching or staining was observed.

The structural engineer who performed the inspection concluded that the drywell shield walls are structurally adequate to perform their intended functions.

The 2002 inspection report noted that the structural condition of the shield walls was the same as that observed in 1998. Cracks observed are minor and that the walls are adequate to perform their intended functions.

The 2005 inspection report noted that the shield walls are in good/sound condition and capable of performing their intended function. The minor hairline cracks and rust stains are the same as noted in previous inspections.

f) Currently, what is the extent of the elevated temperature region and what is the extent of the cracked region (distribution, length, width of cracks)?
Response: As evidenced by operating experience discussed above, the extent of the elevated temperature region and the extent of the cracked region have not significantly changed since the benchmark report of 1994. Additional minor cracks and stains have been observed since that time. However they were not considered significant to impact the intended function of the drywell shield wall.

g) Has there been a need to conduct re-analysis or make any repairs?
Response: A re-analysis was performed for GPU by ABB Impell Corporation (Report #0037-00196-0) and transmitted to NRC in November 19, 1993 (Ref. 5). There has been no need for repairs.

h) Is the LR commitment under the OCGS SMP greater than, equal to, or less than the condition monitoring activities currently being conducted to satisfy the NRC staff's recommendation?
Response: The LR commitment under the Oyster Creek Structures Monitoring aging management is equal to the condition monitoring activities conducted under the current term to satisfy NRC Staff recommendations.

Item b) was revised to provide additional clarification requested by D. V. Hoang (NRC) on 1/25/2005

Response to follow-up Questions:

(1) Cracking of the drywell shield wall is monitored under the structures monitoring program. The program requires visual inspection of the shield wall for new cracks, crack growth and staining of the concrete as described in reference 4. An engineer with a B.S. degree, or a professional engineer, who has a minimum of five years experience working on nuclear structures, conducts inspections. Acceptance criteria are in accordance with ACI 349, which states that passive cracks less than 0.015 inches are acceptable without further evaluation. This criteria envelopes the Staff recommended 0.02 inches crack width criteria for the drywell shield wall. The current procedure does not specify a numerical value for crack width, rather the procedure relies on qualitative assessment by the qualified engineer to establish if observed cracks meet the guidance in ACI 349 and whether they could impact structural integrity of the wall. Previous inspection results, described in item (e) above, indicate that the cracks are generally hairline that require no repair, and that the cracks have exhibited no significant change over the years. The Structures Monitoring Program (B.1.31) implementing procedure will be enhanced to add the Staff recommended criteria for the drywell shield wall crack width (0.02-inch).

(2) The statement in Ref. 4, GPU Nuclear is developing a program to ensure monitoring of concrete conditions during refueling outage and a formal guideline for performing the monitoring (e.g. visual inspections for crack growth and/or staining of the concrete)." is related to actions planned by GPU to implement a formal monitoring program. This planned formal program has been incorporated into the Structures Monitoring Program as discussed in item (1) above. The normal inspection frequency of the Structures Monitoring Program of 4 years was reduced to every refueling outage for the drywell

shield wall consistent the statement in Ref. 4.

(3) Oyster Creek has assigned a structural system engineer to monitor the condition of the drywell shield wall as well as other structures. The engineer is a licensed professional engineer with a minimum of 5 years experience with nuclear structures. Inspection and acceptance criteria are as discussed above. Inspection frequency is every refueling outage as stated in reference 4. The enhanced procedure will incorporate NRC Staff recommended 0.02 inches crack width acceptance criteria to be used for future inspections.

(4) The conclusion cited in ABB IMPELL Report #03-0370-1341, Section 5.4, conclusion (4) related to the effects of consolidated fuel loads is not implemented at Oyster Creek. The term consolidated fuel refers to removing the hardware from fuel bundles, such as channels and end plates, to allow for storage of greater quantity of spent fuel in the high density racks. The consolidated fuel loads are therefore greater than loads that are a result of the fuel rack change. The ABB IMPELL analysis included the consolidated fuel load in one of the load combinations to determine if the spent fuel pool structure will support it. However as stated above, Oyster Creek does not store fuel in the spent fuel storage pool in a consolidated form.

With respect to the cracking in the spent fuel pool area cited in ABB IMPELL analysis, Oyster Creek has observed cracks on the concrete girder along Column Line RE, and the bottom of the floor slab beneath the spent fuel pool north wall and, the drywell shield wall cracking. The observed cracks were attributed to temperature conditions and little cracking, if any, takes place under sustained loads. As cited in ABB IMPELL Report cracking predicted by the analysis is closely correlated with observed cracking. The analysis showed that the spent fuel pool structure is in full compliance with ACI 349-80 for all loads for which the plant was licensed. The analysis also concluded that the spent fuel pool structure is capable of supporting consolidated fuel load; but the stress margin in certain components is zero. The zero margin in this case is academic since Oyster Creek does not store fuel in a consolidated form.

Subsequently, four additional fuel racks were installed in the year 2000. A finite element analysis of the fuel pool structure was performed by Holtec International and is documented in Holtec Report HI-981983.

Monitoring of the cracks identified in the spent fuel pool area is included in the existing Oyster Creek Structures Monitoring Program (B.1.31). The program is credited for aging management of the cracks during the period of extended operation.

Revised response to add response to "Follow-Up Question". AMO 2/26/06.
Revised Folllow-up Question 4. TEQ 2/28/06

References:

1. NUREG-0822, "Integrated Plant Safety Assessment, Systematic Evaluation Program", Oyster Creek Nuclear Generating Station Final Report Dated January 1983.
2. Letter, J. Zwolinsky (NRC) to P. Fiedler (GPUN) with a Safety Evaluation 4.12 (SEP Topic III-7.B)

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of NUREG-0822, Design Codes, Design Criteria and Load Combinations, Dated Oct. 29, 1986.
3. Letter from Alexander W. Dromerick, Jr. (NRC) to J. Barton (GPU), "Oyster Creek Nuclear Generating station - Evaluation of Effects of High Temperature on drywell Shield Wall and Biological Shield Wall, SEP Topic III-7.B "Design Codes, Design Criteria, Load Combinations, and Reactor Cavity Criteria" (TAC No. M76879) dated May 11, 1994.
4. Letter from R.W. Keaten to U.S. NRC, "Oyster Creek Nuclear Generating Station (OCNGS) Docket 50-219 SEP Topic III-7B, drywell Shield Wall Integrity", dated April 19, 1994
5. Letter, R. Keaton (GPUN) to NRC, Response to Request for Additional Information on Drywell Temperature (SEP Topic III-7.B), Dated November 19, 1993.
6. Holtec Report HI-981983, "Licensing Report for Storage Capacity Expansion of Oyster Creek Spent Fuel Pool" ,Revision 4, dated June 15, 1999

***LRCR #:*** 277 ***LRA A.5 Commitment #:***

***IR#:***

***Approvals:***

***Prepared By:*** Ouaou, Ahmed 2/26/2006

***Reviewed By:*** Quintenz, Tom 2/28/2006

***Approved By:*** Warfel, Don 2/28/2006

***NRC Acceptance (Date):***

# NRC Information Request Form

*Item No*
AMR-355

*Date Received:* 2/16/2006
*Source* AMR Audit

*Status:* Open

*Topic:*
Aging Management of Auxiliary Systems

*Document References:*
3.3

*NRC Representative* Lofaro, Bob

*AmerGen (Took Issue):* Hufnagel, Joh

***Question***

Question 3.3-15
In the OCGS document titled Reconciliation of Program and Line Item Differences Between January 2005 Draft NUREG-1801 and September 2005 NUREG-1801, Section 1 states that It is the OCGS license renewal team's understanding of the NRC's expectation of the scope of reconciliation that new line items added in September 2005 Revision 1NUREG-1801 do not have to be considered in this reconciliation. Please clarify the intent of this statement and how it impacts the reconciliation performed.

*Assigned To:* Getz, Stu

***Response:***

The Reconciliation document made available to the NRC Audit Team on January 23, 2006 reflected the Oyster Creek License Renewal Team's understanding that the NRC's expectation of the scope of reconciliation did not include evaluation of new line items added to the September 2005 Revision 1 NUREG-1800 and 1801 documents. The January 2005 draft SRP and GALL documents used to generate the Oyster Creek LRA did not have 88 Table 2 line items that were added in the September 2005 Revision 1 version of those documents.

The Oyster Creek License Renewal Team has subsequently performed an evaluation of those new line items to determine their applicability to Oyster Creek Generating Station. The results of this evaluation will be included in a new attachment to the Reconciliation document, and will be made available to the NRC for its review in a manner to be agreed upon between AmerGen and the License Renewal Project Manager and Audit Team Lead.

*LRCR #:* 278
*LRA A.5 Commitment #:*

*IR#:*

***Approvals:***

*Prepared By:* Getz, Stu 2/27/2006

# *NRC Information Request Form*

***Reviewed By:*** Beck, George 3/ 1/2006

***Approved By:*** Polaski, Fred 3/ 2/2006

***NRC Acceptance (Date):***