

January 19, 2005

LICENSEE: Nuclear Management Company, LLC

FACILITY: Point Beach Nuclear Plant, Units 1 and 2

SUBJECT: SUMMARY OF TELEPHONE CONFERENCE HELD ON JANUARY 10, 2005, BETWEEN THE U.S. NUCLEAR REGULATORY COMMISSION AND NUCLEAR MANAGEMENT COMPANY, LLC, CONCERNING REQUESTS FOR ADDITIONAL INFORMATION PERTAINING TO THE POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2, LICENSE RENEWAL APPLICATION

The U.S. Nuclear Regulatory Commission staff (the staff) and representatives of Nuclear Management Company, LLC (NMC) held a telephone conference on January 10, 2005, to discuss and clarify the staff's requests for additional information (RAIs) concerning the Point Beach Nuclear Plant, Units 1 and 2, license renewal application. The conference call was useful in clarifying the intent of the staff's RAIs.

Enclosure 1 provides a listing of the meeting participants. Enclosure 2 contains a listing of the RAIs discussed with the applicant, including a brief description on the status of the items. Enclosure 3 contains draft responses provided by the applicant.

The applicant had an opportunity to comment on this summary.

/RA/

Verónica M. Rodríguez, Project Manager
License Renewal Section A
License Renewal and Environmental Impacts Program
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Docket Nos. 50-266 and 50-301

Enclosures: As stated

cc w/encls: See next page

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SUBJECT: SUMMARY OF TELEPHONE CONFERENCE HELD ON JANUARY 10, 2005, BETWEEN THE U.S. NUCLEAR REGULATORY COMMISSION AND NUCLEAR MANAGEMENT COMPANY, LLC, CONCERNING REQUESTS FOR ADDITIONAL INFORMATION PERTAINING TO THE POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2, LICENSE RENEWAL APPLICATION

The U.S. Nuclear Regulatory Commission staff (the staff) and representatives of Nuclear Management Company, LLC (NMC) held a telephone conference on January 10, 2005, to discuss and clarify the staff's requests for additional information (RAIs) concerning the Point Beach Nuclear Plant, Units 1 and 2, license renewal application. The conference call was useful in clarifying the intent of the staff's RAIs.

Enclosure 1 provides a listing of the meeting participants. Enclosure 2 contains a listing of the RAIs discussed with the applicant, including a brief description on the status of the items. Enclosure 3 contains draft responses provided by the applicant.

The applicant had an opportunity to comment on this summary.

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ADAMS Accession No.: **ML050260247**

DOCUMENT NAME: E:\Filenet\ML050260247.wpd

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DISTRIBUTION: Note to Licensee: Nuclear Management Company,LLC, Re: Summary of Telephone Conference Held on January 10,2005, Dated: January 19, 2005
Adams Accession No.: **ML050260247**

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TO DISCUSS THE POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2
LICENSE RENEWAL APPLICATION
JANUARY 10, 2005

Participants

Affiliations

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DRAFT REQUESTS FOR ADDITIONAL INFORMATION (RAI)
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2
LICENSE RENEWAL APPLICATION

January 10, 2005

The U.S. Nuclear Regulatory Commission staff (the staff) and representatives of Nuclear Management Company, LLC (NMC) held a telephone conference call on January 10, 2005, to discuss and clarify the staff's requests for additional information (RAIs) concerning the Point Beach Nuclear Plant, Units 1 and 2, license renewal application (LRA). The following RAIs were discussed during the telephone conference call.

Aging Management Review for Reactor Coolant System and Pressurizer

RAI 3.1.2-1

In your LRA, you indicated that WCAP-14575-A and 14574-A was approved by the staff. Please indicate the dates of the approval letters/safety evaluation for the subject WCAPs.

Discussion: The applicant clarified their draft response. The applicant will provide their formal response in writing.

RAI 3.1.2-2

In your LRA application, you indicated that meaningful volumetric inspection techniques did not exist for socket welds in the Class 1 piping. In light of the successful application of UT of socket welded joints in one inch piping at Susquehanna Steam Electric Station, please discuss the applicability for this technique at PBNP Units 1 and 2 to manage the aging effects of fatigue. If you determine that this application is not suitable for your plant, the discussion should describe in detail why this technique is not viable at PBNP Units 1 and 2, and the basis why no safety significant condition exists if not implemented.

Discussion: The applicant clarified their draft response. The applicant will provide their formal response in writing.

RAI 3.1.2-3

In your plant specific response, item (10), Table 3.1.0-1, you indicated that plant process control procedures (design control, repair/replacement, and welding) will be revised to ensure that repair or replacement of Class 1 piping components welded connections or cast austenitic stainless steel (CASS) would require a new LBB analysis based on replacement process and/or material properties. Prior to that statement, you indicated that the subject LBB analyses had been revised addressing SG replacement, power uprate, and a 60-year operating period. Since these LBB analyses revisions typically address the effects of thermal aging on CASS components, please explain in detail why the revisions to the procedures are necessary. In your response, please advise if the revised LBB analyses address the effects of thermal aging on CASS components.

Enclosure 2

Secondly, if the revisions to the plant control procedures are required per your plant administrative program controls, please provide a commitment to your LRA that the subject revisions will be completed prior to the period of extended operation.

Discussion: The applicant clarified their draft response. The applicant will provide their formal response in writing.

RAI 3.1.2-4

Under the plant-specific response in Table 3.1.0-3 for Renewal Applicant Action Item (4), you stated that absolute assurance could not be provided that the yield strength of your SA-193, Grade B7 bolting is under 150 ksi. Furthermore, you stated that since the Inservice Inspection database results show that no cracking is occurring, you do not consider SCC an aging effect requiring management for the Point Beach pressurizer bolting. Please explain in detail, the type, extent, and frequency of nondestructive examinations on this pressure retaining bolting performed at Point Beach under your Inservice Inspection Program.

Discussion: The applicant clarified their draft response. The applicant will provide their formal response in writing.

RAI 3.1.2-5

In Tables 3.1.2-1 and 3.1.2-4, you indicate that the Water Chemistry Control AMP is used to manage the effects of loss of material due to corrosion in low flow and stagnant areas for a variety of stainless and cast stainless materials. These components are designated by Note H with footnotes 5 and 21. NUREG-1801, XI.M2 recommends a One-Time Inspection Program to validate the effectiveness of the Chemistry Control Program for low flow and stagnant areas because the mitigating effects of a Water Chemistry Control program are effective for intermediate and high-flow areas. Please discuss how the aging effect of loss of material for components specified under Note H of the two listed tables is managed for low flow/stagnant areas since NUREG-1801, XI.M2 specifies that the mitigating effects of a Chemistry Control Program alone is not sufficient.

Discussion: The applicant clarified their draft response. The applicant will provide their formal response in writing.

Alloy 600 Program RAI

RAI 2.1.16-1

Please provide a commitment to assure that interim report "PWR Materials Reliability Project Interim Alloy 600 Safety Assessment for US PWR Plants (MRP-44), Part 1: Alloy 82/182 Pipe Butt Welds," and its final version will be used as part of the basis for the Reactor Coolant System Alloy 600 Inspection Program. The commitment should state that the Reactor Coolant System Alloy 600 Inspection Program will be submitted 24 - 36 months prior to the period of extended operation for staff review and approval to determine if the program demonstrates the ability to manage the effects of aging per 10 CFR 50.54.21(a)(3).

Discussion: The applicant clarified their draft response. The applicant will provide their formal response in writing.

RAI 2.1.16-2

Please discuss in detail your review of industry/plant operating experience and how it will equate to the continued operation of the existing PBNP Units 1 and 2 RPV heads. If the heads are to be replaced, please discuss your plans for the monitoring of the heads in accordance with current industry guidance, Owner's Groups activities and existing NRC regulations or Orders.

Discussion: The applicant clarified their draft response. The applicant will provide their formal response in writing.

Leak Before Break TLAA

RAI 4.1.1-1

Please discuss whether there are any calculations or analyses at PBNP that address the topics listed in 4.1.1.1 of the application and were not included in Table 4.1-2 of the LRA.

Discussion: The applicant clarified their draft response. The applicant will provide their formal response in writing.

RAI 4.1.1-2

In section 4.1.2, pursuant to 10 CFR 54.21(c)(2), a list of plant specific exemptions granted pursuant to 10 CFR 50.12 was provided. This section described the exemptions and why they were still needed. Pursuant to 10 CFR 54.21.(c)(2), please provide an evaluation that justifies the continuation of these exemptions for the period of extended operation.

Discussion: The applicant clarified their draft response. The applicant will provide their formal response in writing.

ENCLOSURE

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2 LICENSE RENEWAL APPLICATION

NRC Question RAI 3.1.2.-1 (Aging Management Review for Reactor Coolant System and Pressurizer):

In your LRA, you indicated that WCAP-14575-A and 14574-A was approved by the staff. Please indicate the dates of the approval letters/safety evaluation for the subject WCAPs.

NMC Response:

The approval letter for WCAP-14574-A, "Aging Management Evaluation for Pressurizers", was dated October 26, 2000. The safety evaluation was an enclosure to the referenced approval letter, and was not dated.

The approval letter for WCAP-14575-A, "Aging Management Evaluation for Class 1 Piping and Associated Pressure Boundary Components", was dated November 8, 2000. The safety evaluation was an enclosure to the referenced approval letter, and was not dated.

NRC Question RAI 3.1.2.-2 (Aging Management Review for Reactor Coolant System and Pressurizer):

In your LRA application, you indicated that meaningful volumetric inspection techniques did not exist for socket welds in the Class 1 piping. In light of the successful application of UT of socket welded joints in one inch piping at Susquehanna Steam Electric Station, please discuss the applicability for this technique at PBNP Units 1 and 2 to manage the aging effects of fatigue. If you determine that this application is not suitable for your plant, the discussion should describe in detail why this technique is not viable at PBNP Units 1 and 2, and the basis why no safety significant condition exists if not implemented.

NMC Response:

The Susquehanna Steam Electric Station enhanced the Diablo Canyon UT inspection technique for small bore (3/4 and 1-inch) socket welds. The "Susquehanna" UT technique was intended to inspect socket welds for early detection of ID fatigue cracking to eliminate unscheduled shutdown costs associated with repair of socket weld failures. The "Susquehanna" UT technique has been successfully applied at the Susquehanna Steam Electric Station to identify questionable socket welds, allow repair, and thus reduce vibrational induced fatigue failures of socket welded piping connections.

Only a limited amount of follow up destructive examinations were performed on questionable socket welds identified by the "Susquehanna" UT technique. Two follow up destructive examinations of questionable welds verified a through wall crack, and a 40% through wall crack.

The "Susquehanna" UT technique is only considered a go / no go inspection when an indication is discovered. Characterization of the indication is not possible. The "Susquehanna" UT technique can detect 25 percent through wall defects with a high probability. The "Susquehanna" UT technique can also detect much smaller defects, however the technical limits have not been determined.

The Susquehanna UT technique shows promise in situations dealing with frequent fatigue related socket welded joint failures. The technique is not a Code acceptable inspection, as it does not meet the inspection requirements of ASME for a volumetric examination. The Susquehanna UT technique has not been thoroughly investigated, nor accepted by the industry. In addition, since the Susquehanna UT technique cannot characterize indications, application of the technique can result in rejecting socket welds that may otherwise be acceptable, as demonstrated by application of the technique at the Susquehanna Steam Electric Station.

PBNP has not experienced a high number of socket weld fatigue failures. The failures to date have been primarily associated with isolated original installation defects, or system design modifications.

Three (3) 2-inch socket weld cracks were experienced in the AFW pump recirculation lines downstream of the pressure reducing orifices, following a system modification to allow an increase in the pump recirculation flow rates. The cracks were attributed to vibration induced fatigue originating from the pressure reducing orifices. The welds were repaired and the pressure reducing orifices were replaced to eliminate the source of vibration. Each of the cracks resulted in minor leakage that was detected by plant personal during area inspections.

One (1) 3/4-inch socket weld crack was experienced in a drain line connection in the Safety Injection / Containment Spray systems full flow test line. The full flow test line was a system modification to accommodate minimizing operation of safety grade pumps at minimum flow rates. The failure was attributed to vibration induced fatigue originating from the pressure reducing orifice. The weld was repaired and the piping system support scheme was modified to reduce the vibration. The crack resulted in minor leakage. Since the line is normally isolated from the noted systems, the leakage was detected by operations personal during the performance of the full flow system test.

Two (2) 3/8-inch socket weld cracks were experienced on the S/G channel head drain line isolation valve. The S/G channel head drain lines were included with the replacement Unit 1 S/Gs. The cracks were attributed to fatigue originating from either thermal cycling or the acoustic behavior of the specific geometry and flow conditions. The welds were repaired and increased in size per EPRI guidance. The cracks did not result in detectable leakage, however did leave boric acid evidence that was identified during routine refueling outage inspections.

The small bore piping locations selected by the PBNP Risk-Informed Inservice Inspection (RI-ISI) program are in primary systems, and are located within the containment building. Application of the "Susquehanna" UT inspection technique for socket-welds can easily result in the unnecessary replacement of socket-welded joints in a radiation environment.

The PBNP RI-ISI program was created based on the guidance of EPRI TR-112657, Revision B-A, "Revised Risk-Informed Inservice Inspection Evaluation Procedure". The EPRI TR-112657 RI-ISI methodology specifically considers a variety of aging degradation mechanisms, including fatigue. Both the PBNP RI-ISI program and EPRI TR-112657, Revision B-A, were reviewed and approved by the NRC.

EPRI TR-112657, Revision B-A, notes the following: "The real measure of protection against catastrophic failure of a piping system component is the combination of good design and leak-before-break properties. All of the service induced failure mechanisms which effect nuclear power plant piping except one (flow accelerated corrosion) have been shown to be of a gradually progressing nature, which inevitably produce detectable leakage before significantly reducing the inherent safety margins of the piping relative to gross rupture. The combination of periodic leak tests required by Section XI, in conjunction with continuous leakage monitoring requirements for all primary coolant systems during operation has proven to be more than adequate protection against a large pipe break. The potential for flow accelerated corrosion, which has caused large pipe breaks without prior leakage, is minimal in Class 1 systems."

In addition, the NRC Safety Evaluation of the PBNP RI-ISI program, dated July 2, 2003 notes the following: "The objective of ISI required by ASME Code, Section XI, is to identify conditions (i.e., flaw indications) that are precursors to leaks and ruptures in the pressure boundary that may impact plant safety. The RI-ISI program is judged to meet this objective."

"The methodology used by the licensee also considers implementation and performance monitoring strategies. Inspection strategies ensure that failure mechanisms of concern have been addressed and there is adequate assurance of detecting damage before structural integrity is affected. The risk significance of piping segments is taken into account in defining the inspection scope for the RI-ISI program."

In view that PBNP has only experienced a limited amount of socket weld fatigue failures primarily related to design issues, the Susquehanna UT technique can lead to unnecessary radiation exposure in repairing good socket weld joints, industry experience indicates that fatigue degradation will lead to detectable leakage well before a loss of structural integrity, and the existing RI-ISI program has been found adequate to detect damage before structural integrity is affected, PBNP does not believe that performing the "Susquehanna" UT inspection technique on socket welds at PBNP is justified.

The NMC will actively participate in the EPRI sponsored Materials Reliability Project (MRP) Industry Task Group (ITG) on thermal fatigue. In addition, NMC commits to monitor on-going industry activities related to failure mechanisms for small-bore piping, and will evaluate changes to PBNP inspection activities based on industry recommendations.

NRC Question RAI 3.1.2.-3 (Aging Management Review for Reactor Coolant System and Pressurizer):

In your plant specific response, item (10), Table 3.1.0-1, you indicated that plant process control procedures (design control, repair/replacement, and welding) will be revised to ensure that repair or replacement of Class 1 piping components welded connections or cast austenitic stainless steel (CASS) would require a new LBB analysis based on replacement process and/or material properties. Prior to that statement, you indicated that the subject LBB analyses had been revised addressing SG replacement, power uprate, and a 60-year operating period. Since these LBB analyses revisions typically address the effects of thermal aging on CASS components, please explain in detail why the revisions to the procedures are necessary. In your response, please advise if the revised LBB analyses address the effects of thermal aging on CASS components.

Secondly, if the revisions to the plant control procedures are required per your plant administrative program controls, please provide a commitment to your LRA that the subject revisions will be completed prior to the period of extended operation.

NMC Response:

The PBNP LBB analyses have been revised to address Steam Generator Replacement, Power Uprate, and a 60-year operating period. The revised LBB analyses account for the effects of thermal aging of CASS materials.

Applicant Action Item 11 of the NRC Safety Evaluation for WCAP-14575-A, Aging Management Evaluation for Class 1 Piping and Associated Pressure Boundary Components”, requires that provisions for management of Thermal Aging (LBB Analyses) include addressing the impacts of repairs/replacements of CASS components on the LBB analyses. This is the reason the proposed revisions to the noted PBNP procedures/processes were deemed necessary.

In NMC Letter to the NRC, NRC 2004-0016, dated February 25, 2004, PBNP has committed to implement an enhanced ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program prior to the period of extended operation. The LRA, Appendix B, Subsection B2.1.1 identifies the necessary program enhancements. Revisions to the noted procedures/processes to address maintenance of the LBB analyses are included in the referenced LRA section. Thus, PBNP has already committed to performing the subject enhancements.

NRC Question RAI 3.1.2.-4 (Aging Management Review for Reactor Coolant System and Pressurizer):

Under the plant-specific response in Table 3.1.0-3 for Renewal Applicant Action Item (4), you stated that absolute assurance could not be provided that the yield strength of your SA-193, Grade B7 bolting is under 150 ksi. Furthermore, you stated that since the Inservice Inspection database results show that no cracking is occurring, you do not consider SCC an aging effect requiring management for the Point Beach pressurizer bolting. Please explain in detail, the type, extent, and frequency of nondestructive examinations on this pressure retaining bolting performed at Point Beach under your Inservice Inspection Program.

NMC Response:

The pressurizer bolting is inspected in accordance with ASME Section XI, Table IWB-2500-1, Category B-G-2, Item No. B7.20. The inspection is a VT-1 (surface visual) of 100% of the bolting, performed once per interval.

In addition to the ASME Section XI bolting inspections, the pressurizer bolting is visually inspected by trained maintenance personnel each time the bolting is removed for maintenance activities. These additional inspections are performed as a result of PBNP's response to NRC IEB 82-02, "Degradation of Threaded Fasteners in the Reactor Coolant Pressure Boundary of PWR Plants". Since the pressurizer bolting at PBNP is removed, cleaned, and re-torqued on a refueling basis (~18 months), the maintenance inspection is also performed on a refueling basis.

NRC Question RAI 3.1.2.-5 (Aging Management Review for Reactor Coolant System and Pressurizer):

In Tables 3.1.2-1 and 3.1.2-4, you indicate that the Water Chemistry Control AMP is used to manage the effects of loss of material due to corrosion in low flow and stagnant areas for a variety of stainless and cast stainless materials. These components are designated by Note H with footnotes 5 and 21. NUREG-1801, XI.M2 recommends a One-Time Inspection Program to validate the effectiveness of the Chemistry Control Program for low flow and stagnant areas because the mitigating effects of a Water Chemistry Control program are effective for intermediate and high-flow areas. Please discuss how the aging effect of loss of material for components specified under Note H of the two listed tables is managed for low flow/stagnant areas since NUREG-1801, XI.M2 specifies that the mitigating effects of a Chemistry Control Program alone is not sufficient.

NMC Response:

In NUREG-1801, Chapter IV, Section C2, REACTOR COOLANT SYSTEM AND CONNECTED LINES (PRESSURIZED WATER REACTOR), none of the line items corresponding to Class 1 piping components or the pressurizer, refer to loss of material due to corrosion, they only refer to cracking. (This condition is reflected by the use of Note H, which is simply intended to communicate that the aging effect of loss of material was not included in the GALL tables for these components.) The reason GALL did not identify this aging effect may be because loss of material due to corrosion was considered to be non-significant/not susceptible in a PWR primary coolant environment.* However, this is due to the fact that the PWR primary coolant environment is strictly controlled by the Water Chemistry Program, and therefore PBNP conservatively included this aging effect in Tables 3.1.2-1 and 3.1.2-4 to be managed by the Water Chemistry Program. Note that every component identified with this aging effect also identifies cracking as a separate aging effect, that is managed in-part by the Inservice Inspection Program. While these inservice inspections will look primarily for cracking, they can identify loss of material due to corrosion also. Plant-specific and industry operating experience has shown the loss of material due to corrosion is not actively occurring in PWR primary coolant environments. This is based on many internal inspections that have been (and will continue to be) performed on Class 1 piping and components.

For these reasons, additional one-time inspections are not warranted for the management of loss of material due to corrosion, for components in PBNP's Class 1 piping/components and pressurizer.

* (See NRC FSER for WCAP-14574, Aging Management Evaluation for Pressurizers, p. 21-22, along with Applicant Action Item 3.2.2.1-1; and NRC FSER for WCAP-14575, Aging Management Evaluation for Class 1 Piping and Associated Pressure Boundary Components, p. 17.)

NRC Question RAI 2.1.16.-1 (Alloy 600 Program):

Please provide a commitment to assure that interim report "PWR Materials Reliability Project Interim Alloy 600 Safety Assessment for US PWR Plants (MRP-44), Part 1: Alloy 82/182 Pipe Butt Welds," and its final version will be used as part of the basis for the Reactor Coolant System Alloy 600 Inspection Program. The commitment should state that the Reactor Coolant System Alloy 600 Inspection Program will be submitted 24 - 36 months prior to the period of extended operation for staff review and approval to determine if the program demonstrates the ability to manage the effects of aging per 10 CFR 50.54.21(a)(3).

NMC Response:

PBNP will commit to use of the interim report "PWR Materials Reliability Project Interim Alloy 600 Safety Assessment for US PWR Plants (MRP-44), Part 1: Alloy 82/182 Pipe Butt Welds," and its final version as part of the basis for the Reactor Coolant System Alloy 600 Inspection Program. PBNP further commits to submittal of the Reactor Coolant System Alloy 600 Inspection Program to the NRC for staff review and approval 24 - 36 months prior to the period of extended operation.

NRC Question RAI 2.1.16.-2 (Alloy 600 Program):

Please discuss in detail your review of industry/plant operating experience and how it will equate to the continued operation of the existing PBNP Units 1 and 2 RPV heads. If the heads are to be replaced, please discuss your plans for the monitoring of the heads in accordance with current industry guidance, Owner's Groups activities and existing NRC regulations or Orders.

NMC Response:

The PBNP RPV Heads will be replaced during each Unit's upcoming refueling outage in 2005.

The replacement PBNP RPV Heads will be inspected in accordance with the requirements of NRC Order EA-03-009, "Order Establishing Interim Inspection Requirements for Reactor Pressure Vessel Heads at Pressurized Water Reactors," revised February 20, 2004.

The NMC is actively participating with the industry through the EPRI MRP efforts to develop long term inspection requirements for reactor vessel closure heads and their penetrations for U.S. pressurized water reactor plants.

NRC Question RAI 4.1.1-1 (Leak Before Break TLAA):

Please discuss whether there are any calculations or analyses at PBNP that address the topics listed in 4.1.1.1 of the application and were not included in Table 4.1-2 of the LRA.

NMC Response:

There are no calculations or analyses at PBNP that address the topics listed in Section 4.1.1.1 of the application that were not included in Table 4.1-2 of the LRA.

NRC Question RAI 4.1.1-2 (Leak Before Break TLAA):

In section 4.1.2, pursuant to 10 CFR 54.21(c)(2), a list of plant specific exemptions granted pursuant to 10 CFR 50.12 was provided. This section described the exemptions and why they were still needed. Pursuant to 10 CFR 54.21.(c)(2), please provide an evaluation that justifies the continuation of these exemptions for the period of extended operation.

NMC Response:

The NMC withdrew the request for exemptions to 10 CFR 50.61, and Appendices G and H to 10 CFR 50 (Reference Letter from NMC to NRC dated August 3, 2004, NRC 2004-0079). Revised sections of the LRA were included in NMC Letter to the NRC dated September 10, 2004 (NRC 2004-0085). The revised Section 4.1.2 now states, "No TLAA related exemptions granted pursuant to 10 CFR 50.12 were identified."