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RBG-46037

December 20, 2002

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

SUBJECT: River Bend Station, Unit 1
Docket No. 50-458
Response to Request for Additional Information
One-time Extension of the ILRT Interval
License Amendment Request (LAR) 2002-16

REFERENCE: Entergy letter dated May 14, 2002, License Amendment Request 2002-16, "One-time Extension of the Integrated Leak Rate Test Interval" (RBG-45952)

Dear Sir or Madam:

Entergy Operations, Inc. (Entergy), in Reference 1, requested approval of changes to the River Bend Station, Unit 1 Technical Specifications associated with extending the allowed interval between the integrated leak rate tests (ILRT). Based on their initial review, the NRC has provided a request for additional information (RAI) regarding this application. Responses to the questions are provided in Attachment 1.

There are no new commitments made in this letter. Should you have any questions or comments concerning this request, please contact Jerry Burford at (601) 368-5755.

I declare under penalty of perjury that the foregoing is true and correct. Executed on December 20, 2002.

Sincerely,

PDH/FGB

A017

Attachment:
1. Response to RAI

cc: U. S. Nuclear Regulatory Commission
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NRC Senior Resident Inspector
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Attachment 1

RBG-46037

Response to Request for Additional Information

Response to Request for Additional Information related to request to extend the ILRT interval

As the inservice inspection requirements mandated by 10 CFR 50.55a and the leak rate testing requirements of Option B of Appendix J complement each other in ensuring the leak-tight and structural integrity of the containment, the staff needs the following information to complete its review of the license amendment request.

1. In discussing Item 1 of the Containment Inservice Inspection (ISI) program, the licensee describes the visual examinations of the containment surfaces (sometimes one surface, sometimes both of the surfaces). The licensee is requested to provide a summary of the screening criteria used in CEP-CII-006, and a summary (including location, size, root cause, etc.) of the degradations found during these examinations, and actions (corrective or analytical evaluation) taken when the degradation exceeded 10% of the shell thickness.

Response:

The program plan CEP-CII-006 is the controlling document for the overall containment inspection program and references other procedures as necessary to control specific examinations. RBS procedure STP-057-3700 currently controls general visual examinations conducted at RBS and delineates acceptance criteria for these examinations. As a part of ongoing standardization efforts, it is anticipated that STP-057-3700 will be superceded by CEP-CII-003 prior to the next set of examinations and have screening criteria similar to those established for Waterford 3, Grand Gulf Nuclear Station, and Arkansas Nuclear One.

The general visual examinations conducted to date at River Bend have been conducted in accordance with site procedure STP-057-3700. The acceptance criteria for examinations conducted at River Bend Station is provided in Section 8 of STP-057-3700 and repeated below:

For Painting and Coating:

- Blistering greater than size No. 6 as specified in ASTM D714
- Checking greater than standard No. 2 as specified in ASTM D 660
- Cracking greater than standard No. 6 as specified in ASTM D 661
- Flaking greater than standard No. 6 as specified in ASTM D 772
- Rusting equal to or greater than Grade 7 as specified in ASTM D 610
- Other distress that may indicate degradation of the base metal as determined by 10CFR50 Appendix J or the RPE/RE

For Class MC (the vessel or liner itself)

- Excessive Corrosion (Defined in the definitions section)
- Deep gouges or dents (excluding fabrication and installation marks)
- Excessive wear (generally represented by shiny surfaces, ridges, evidence of motion, or other material wastage)
- Bulging

- Other damage, deformation, or degradation, such as cracks, arc strikes, tears, broken welds, etc., that is is considered by the 10CFR50 Appendix J coordinator or the RPE/RE as harmful to the material condition within the examination boundary.

While examinations have identified initially suspected areas, all identified areas were determined to be within the acceptance criteria and acceptable as is. No areas with greater than 10% wall thickness loss were noted. Additionally, the areas which were initially considered suspect were cleaned to remove light surface corrosion and any damaged coating and then recoated.

The screening criteria listed in CEP-CII-003 for Waterford 3 provides an example of the how revised screening criteria may look once the standardized program is adopted. These screening criteria are to be developed by the RPE for containment inspection. The Waterford 3 criteria are as follows:

Uncoated Surface Areas

If any of the conditions listed below are present, the condition must be recorded on the General Visual Examination Report and forwarded to the RE for acceptance review:

- Cracking in the base metal.
- Discoloration in the following categories:
    - Visible rust streaks or runs when the source is inaccessible for examination.
    - Significant discoloration of unknown origin.
    - Discoloration of known origin when the origin could have significant impact on the structural integrity or leak tightness of the containment vessel.
- Excessive wear which results discernable thinning of the base metal.
- Pits, dents, or gouges (excluding fabrication marks) of the base metal exceeding 0.09 inches in depth.
- Excessive corrosion as defined in Section 3 of this program section.
- Discernable bulges
- Other conditions causing discernable degradation of the base metal.

Coated Surface Areas

If any of the conditions listed below are present, the condition must be recorded on the examination form and form forwarded to the RE for acceptance review:

- Any of the conditions listed in section 1.1 for uncoated surfaces,
- Uncoated surfaces when the surface is required to be coated.
- Blisters greater than size No. 6 as specified in ASTM D 714 (Annual Book of ASTM Standard, Part 27).
- Excessive Wear defined as wear which results in removal of the coating material to expose bare metal.
- Other distress to the coating that may indicate degradation of the base metal.

It should be noted that the specific requirements for RBS in CEP-CII-003 will be based on RBS design and may have some subtle differences. For instance, it is anticipated that the existing STP criteria for checking of coatings will be retained and that the allowable thicknesses of dents or gouges will vary (to be no greater than 10% of the design wall thickness.).

2. In responding to the NRC's potential question on areas of augmented inspection (as per IWE-1240), the licensee states, "There are currently no augmented examination required for RBS." The licensee is requested to provide information related to the inspection of bottom liner plate and embedments of the RBS containment submerged in water. Specifically, the licensee is requested to provide information regarding the frequency of inspection, and results of the last inspection.

Response:

In accordance with IWE Table 2500-1 (ASME B&PV Code Sub Section IWE 1992 Ed, w/ 92 addenda), wetted surfaces of submerged areas are examined during the VT-3 examination required to be conducted at the end of the inspection interval. Since the window for these examinations at end of the first interval requiring IWE inspections has not yet opened, no examinations of the wetted surfaces have been completed under the Containment Inservice Inspection Program.

3. In responding to the NRC's potential question on examination of seals and gaskets and bolting of the containment pressure retaining components, the licensee refers to the relief authorization granted by NRC, and states, "the Type B testing will be performed at least once during each Containment Inspection Interval."

   Recognizing the hardship associated with examining these components during each inspection period, and that the examination will be performed prior to Type B testing as required by Option A of Appendix J, the staff had granted such relief to a number of licensees. However, implementation of Option B of Appendix J allows flexibility in performing Type B testing based on the leak rate performance of the penetrations. As the performance based testing allows certain leak rate through the penetrations, minor initial degradation of the associated seals, gaskets and bolting can go undetected, and 10 year examination interval could be too long for the degraded components. Thus, examination of seals, gaskets and pressure retaining bolting should be scheduled based on their performance (i.e., plant-specific experience, replacement schedules for resilient seals, etc.), to ensure that, if Type B testing is not performed during the ILRT extension period, the examination schedule will detect degradation of these components. In view of this discussion please provide a schedule for examining (testing) of these components for equipment hatches and other penetrations with resilient seals.

Response:

Type B tests for the following components have been and are performed once every cycle (18 months) after re-assembly. One past exception was one short cycle (Cycle 9, which was only 9 months long), when the Type B test was performed once in 30 months.

- Containment Equipment Hatch
- Control Rod Drive Removal Hatch
- Inclined Fuel Transfer Tube (IFTS) - The IFTS Blind Flange O-Rings are replaced once every cycle

Type B tests for the Containment Airlocks are performed once every 30 months.

4. In responding to the NRC's potential question on containment penetration expansion bellows, the licensee has provided the following information (Ref. 1):

   - 20 penetrations with expansion bellows;
   - Provisions for performing leakage rate tests in-between the two plies of the bellows;
   - Administrative allowable leak rate of 20 Standard Cubic Centimeter Minute (SCCM) has been set for 18 penetration bellows;
   - Two bellows have been found with minor flaws, the allowable administrative leakage rate limit on these bellows have been increased to 65 SCCM, and 88 SCCM.

   For the bellows with minor flaws, the licensee is requested to provide the following information:

   - How often these bellows as well as other 18 bellows are tested?
   - What has been found the root cause for the higher leakage rate through these bellows?
   - How far the administrative limits will be increased before the licensee decides to repair or replace these bellows?

Response:

The bellows with the minor flaws are KJB-Z19 and KJB-Z20, and these bellows failed their original administrative limit of 20 sccm in 1992 (during RF4). The leakage of these bellows was monitored after the flaws were identified; the leakage was found to be low and constant. The measured leak rates were:

KJB-Z19: 34.5 sccm (RF5), 18.0 (RF6), 18.4 (RF7) and 18.2 (RF8)
KJB-Z20: 37.3 sccm (RF5), 33.0 (RF6), 39.1 (RF7) and 35.2 (RF8)

It was determined that the bellows had minor flaws and the administrative limits were re-established at 65 sccm for KJB-Z19 and 88 sccm KJB-Z20. A modification request was initiated to replace the bellows but was cancelled later due to low leakage rates and level trend of the leakage.

Both KJB-Z19 and KJB-Z20 have passed tests to their revised administrative limits in two consecutive LLRT tests (RF7 and RF8). The test frequency has now been extended to 60 months. The next tests of these bellows are scheduled for the upcoming refueling outage (RF11, in March 2003) for both bellows.

All of the other 18 bellows are tested on a 10-yr frequency. Some of them are also scheduled to be tested during RF11.

5. Inspections of some reinforced and steel containments (e.g., North Anna, Brunswick, D. C. Cook, Oyster Creek) have indicated degradation from the uninspectable side of the liner/steel shell of primary containments. The major uninspectable areas of the Mark III containment include those parts of the steel shell backed by concrete, the basemat liner, and inaccessible areas in the annulus. The licensee is requested to provide information as to how potential leakage due to age related degradation from these uninspectable areas are factored into the risk assessment in support of the requested ILRT interval extension.

Response:

Two events of corrosion that initiated from the non-visible (backside) portion of the containment liner have occurred in the industry. These events are summarized below:

- On September 22, 1999, during a coating inspection at North Anna Unit 2, a small paint blister was observed and noted for later inspection and repair. Preliminary analysis determined this to be a through-wall hole. On September 23, a local leak rate test was performed and was well below the allowable leakage. The corrosion appeared to have initiated from a 4"~4"x6' piece of lumber embedded in the concrete.

  An external inspection of the North Anna Containment Structures was performed in September 2001. This inspection (using the naked eye, binoculars, and a tripod-mounted telescope) found several additional pieces of wood in both Unit 1 and Unit 2 Containments. No liner degradation associated with this wood was discovered.

- On April 27, 1999, during a visual inspection of the Brunswick 2 drywell liner, two through-wall holes and a cluster of five small defects (pits) in the drywell shell were discovered. The through-wall holes were believed to have been started from the coated (visible side). The cluster of defects was caused by a worker's glove embedded in the concrete.

The Containment In-service Inspection (CII) program at RBS is described in Entergy procedure CEP-CII-006. The program requirements include a general visual examination of the containment surfaces each inspection period. The general visual examinations are conducted in accordance with Entergy procedures. Any indications exceeding the screening criteria are provided to a qualified containment engineer who compares the indication to the design requirements of the containment vessel. Any indications that exceed the design requirements are documented in the Corrective Action Program and are dispositioned in accordance with the ASME code requirements. The program currently requires VT examinations of bolted connections. The RBS containment design does not incorporate moisture barriers.

Scheduled in-service inspection (ISI) exams are performed in accordance with the requirements of the ASME Section XI, Subsection IWE, and 10 CFR 50.55a. Currently, the RBS program is designed to the 1992 Edition, with 1992 Addenda of ASME XI. These documents require visual examination of essentially 100% of the accessible area of one side of the containment once per ISI period (three in ten years). This exam is performed and documented by Certified NDE Examiners during the outage and/or before an ILRT.

This exam is performed both directly and remotely, depending upon the accessibility to the various areas. To date, the examinations have been performed for both the inner and outer surfaces (during outages RF8 and RF9) and there have been no recordable indications of liner plate degradation. As noted in the original submittal, there are currently no areas requiring Augmented Inspections.

There are inaccessible areas of the RBS containment, including parts of the inner and outer surfaces covered or blocked by concrete or submerged in the suppression pool. The overall inner surface area of the Reactor Building is approximately 84,440 square feet. Of this area, approximately 20,720 square feet, or about 25%, is either covered by concrete or submerged. Approximately 11,300 square feet, or about 13.4% of the total surface area, is associated with the base mat and containment floor area and is inaccessible due to being covered in concrete. There are no programs that monitor the condition of the inaccessible areas of the containment plate directly. When there is an indication of potential degradation of inaccessible areas of the containment plate, this finding is evaluated and appropriate actions are taken to assure the adequacy and integrity of the containment. Note that no such indications have been detected to date for the RBS containment. The submerged surfaces are accessible and are examined at the end of the Containment Inservice Inspection interval in accordance with the ASME requirements. Please note that the submerged portions are stainless steel clad surfaces.

A simplified analysis of the potential impact of containment corrosion resulting in a leak that would not be detected by the ILRT is provided below. It demonstrates that corrosion is estimated to pose a negligible increase in the risk associated with this requested change.

The following approach was used to determine the change in likelihood, due to extending the ILRT, of detecting containment corrosion. This likelihood was then used to determine the resulting change in risk. The following issues are addressed:

- Differences between the containment base mat, cylinder, and dome;
- The historical flaw likelihood due to concealed corrosion;

- The impact of aging;
- The containment corrosion leakage dependency on containment pressure; and
- The likelihood that visual inspections will detect a flaw.

A simplified analysis of the potential impact of containment corrosion resulting in a leak that would not be detected by the ILRT was provided by Calvert Cliffs Nuclear Power Plant (CCNPP) in their March 27, 2002 supplemental submittal and was also used by ANO-1 in their September 9, 2002 submittal. That analysis appears to be generic in nature and is conservative with respect to RBS (based on the fact that the RBS ILRT pressure is less than that CCNPP). One feature at RBS that contributes to the conservatism of this simplified approach is the RBS suppression pool, which covers the base mat and several feet of liner wall. There would be no gaseous leakage from these submerged portions of the containment. One adjustment that will be made in the methodology is to account for the inaccessible portion of the containment shell area. The result of the analysis, when applied to the adjusted increase in Large Early Release Fraction (LERF) developed above in the discussion of the liner examination, provides a reasonable estimate of the impact for RBS. The inputs and assumptions used in the table below are derived from the CCNPP submittal except where noted.

Table 1
Containment Corrosion Analysis

Step	Description	Shell/Dome Area – Accessible (75%)	Shell Area – Inaccessible (11.6%)	Base mat – Inaccessible (13.4%)
1	Historical Liner Flaw Likelihood	5.2E-03		1.3E-03
2	Age Adjusted Liner Flaw Likelihood (15-year average)	6.27E-03		1.57E-03
3	Increase in Flaw Likelihood between 3 and 15 years	8.7%		2.2%
4	Likelihood of Breach given Liner Flaw	1%	1%	0.1%
5	Visual Examination Detection Failure Likelihood	10%	100%	
6	Likelihood of Non-detected Containment Leakage (step 3 * 4 * 5 * %-accessible)	0.087*0.01*0.1* 0.75 = 0.0065%	0.087*0.01*1.0* 0.116 = 0.010%	0.022*0.001*1.0 *0.134 = 0.0003%

So, based on Table 1 results, the total likelihood of a corrosion-induced, non-detected flaw resulting in containment leakage is the sum of the Step 6 results.

Likelihood of Non-detected Leakage = 0.0065% + 0.010% + 0.0003% = 0.017%

This estimate of a non-detected flaw resulting in containment leakage is very conservative for RBS (and other Boiling Water Reactors with Mark III Containments and a suppression pool) as the pool provides a water seal to protect against gaseous releases. The largest contributor to the likelihood of corrosion-induced leakage is the shell area around the pool.

Given the LERF frequency of 2.28E-7 per year and the total CDF of 9.446E-6 per year, the non-large early release frequency failures for RBS are estimated at 9.2E-6 per year. The total CDF is 9.446E-6/yr. If all of the non-detectable containment leakage events are considered to be LERF, then the increase in LERF associated with the liner corrosion issue is:

Increase in LERF (extension from 3 to 15 years) = 0.017% * 9.2E-6 = 1.6E-9

The total increase in LERF for the extension of the testing frequency, including the impact of the corrosion-induced leakage, can be seen as follows:

3.0E-08 + 1.6E-9 = 3.2E-08

Therefore increasing the ILRT interval to 15 years is considered to be a very small change in LERF in accordance with the risk guidelines of RG 1.174.

Recently, new data of through-wall liner corrosion at D.C. Cook has been identified to the NRC. Note that while this would impact some of the values in Table 1 above, it would not be expected to double the values used in the Calvert Cliffs model. RBS has a free-standing containment; it also has a suppression pool in the bottom of the containment such that even if an undetected liner flaw existed, there would not be any additional gaseous releases due to the water level maintained in the pool. Thus, those features, plus the margin inherent in the risk values presented above, demonstrate that even doubling the likelihood of non-detected leakage would still result in an acceptably small change in LERF. Therefore, the corrosion discovered at D.C. Cook does not affect the conclusion that the ILRT extension is non-risk significant.