From: Wiebe, Joel
Sent: Thursday, May 13, 2010 1:29 PM
To: 'Bill Crouch'; 'Gordon Arent'; 'Tony Langley'
Subject: Watts Bar Unit 2 - Preliminary RAIs for FSAR Sections 5.2.3 through 6.1.1

Requests for Additional Information Related to Watts Bar Unit 2, FSAR Amendment 97, Sections 5.2.3 through 6.1.1, Related to SRP Sections 5.2.3, 5.3.1, 5.3.2, and 6.1.1

RAI 5.2.3-1

Background

In FSAR Section 5.2.3.4, Chemistry of Reactor Coolant, the applicant added a description of the process for zinc addition. The applicant indicated that zinc would be added for the purpose of reducing radionuclide content in the primary system corrosion films, and that the residual zinc content would be maintained at a concentration of 2-8 parts per billion (ppb). FSAR Table 5.2-10 also indicates zinc will be limited to less than 40 ppb during normal power operation.

The staff reviewed several reports documenting industry experience with zinc addition in Pressurized Water Reactors (PWR's), which indicate that there is no concern with crud deposition for plants with low-duty or medium-duty cores (Reference 1, 2), and, in fact, zinc addition typically leads to thinner, more evenly distributed crud on fuel. However, there is currently insufficient operating experience with zinc addition in plants with high-duty cores to be able to conclude that zinc injection would not cause a problem with crud deposition in such plants. Core duty is a measure of the amount of subcooled nucleate boiling (SNB) occurring in the core. Plants with high-duty cores are those with high fluid temperatures and high surface heat flux at the fuel clad causing a portion of the total heat transfer to the coolant to occur by SNB. Although favorable for thermal efficiency, the combination of high temperature and SNB leads to more surface boiling, which is known to enhance the formation of corrosion product deposits (crud) at the cladding surface. The tendency for SNB can be quantified by means of the High Duty Core Index (HDCI), calculated in accordance with Appendix F of Reference 3. Cores with an HDCI of ≥ 150 are considered to be high duty plants, medium duty plants have HDCI of 120-149, and a plant with HDCI ≤ 119 is considered a low-duty plant. Staff calculations based on thermal-hydraulic data from FSAR Chapter 4 indicate the WBNP-2 core may be considered high-duty. There may be alternate methods to determine the amount of SNB other the HDCI, such as detailed thermal hydraulic computer models.

Potential problems with crud deposition could include excessively thick fuel crud, or uneven crud thickness that could lead to crud induced power shift (CIPS), also known as axial offset anomaly. Reference 2 also indicates that fuel clad corrosion cannot be completely ruled out for high-duty cores exposed to zinc addition even though no problems have been observed to date. Reference 2 recommends a fuel surveillance program for high-duty plants implementing zinc addition.

Requested Information

1. Is the WBNP core design considered a high-duty core when the HDCI is calculated in accordance with Appendix F of Reference 3, or an alternate method of evaluation?

2. If the WBNP core is considered high-duty, describe the measures to be taken to ensure that zinc addition does not increase the risk of CIPS and or clad corrosion, and that the overall risk of adverse fuel effects is minimized. Possible measures could include, but are not limited to:

a. Implementation of a fuel surveillance program monitoring crud buildup and clad corrosion;
b. Additional chemistry monitoring
c. Application of operating experience with similar core designs.

RAI 5.3.1-1

Background

The regulatory acceptance criteria for a reactor vessel material surveillance program are the requirements of Section III of Appendix H of 10 CFR Part 50. Complying with the acceptance criteria satisfies the requirements of 10 CFR Part 50, Appendix A, General Design Criteria (GDC) 32 regarding an appropriate material surveillance program for the reactor vessel. 10 CFR Part 50 Appendix H, paragraph III.B.3 requires:

> "A proposed withdrawal schedule must be submitted with a technical justification as specified in § 50.4. The proposed schedule must be approved prior to implementation."

Additionally, Generic Letter 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits," contains seven criteria that must be met for the relocation of the pressure-temperature (P-T) limits from the technical specifications to a pressure-temperature limits report (PTLR). One of these criteria is:

> "The Reactor Vessel Material Surveillance Program shall comply with Appendix H to 10 CFR Part 50. The reactor vessel material irradiation surveillance specimen removal schedule shall be provided, along with how the specimen examinations shall be used to update the PTLR curves."

In Amendment 97, FSAR Section 5.4.3.6 was modified to state that the tentative schedule for removal of the capsules for post-irradiation testing is as shown in Table 4.0-1 of the Pressure and Temperature Limits Reports (PTLER). The actual proposed schedule was deleted from the FSAR. However, the PTLR (WCAP-17035-NP, Reference 4) does not contain this information.

Requested Information

1. Provide the proposed Table 4.0-1 which incorporates the schedule for withdrawal of the surveillance capsules in the PTLR.

2. Include a description of the reactor vessel material surveillance program in the PTLR, including a discussion of how the specimen examinations shall be used to update the P-T curves.

RAI 5.3.1-2

Background

10 CFR Part 50, Appendix H, Paragraph III.B.1 requires that the design of the surveillance program and the withdrawal schedule to meet the requirements of the edition of ASTM E 185 that is current on the issue date of the American Society of Mechanical Engineers (ASME) Code to which the reactor vessel was purchased, or later editions through ASTM E 185-1982.

FSAR Section 5.2.4 indicates that changes in fracture toughness of the core region forgings, weldments and associated heat affected zones (HAZ) due to radiation damage will be monitored by a surveillance program which is based on ASTM E 185-82, (Ref. 5) and 10 CFR Part 50, Appendix H. FSAR Section 5.2.4 further indicates that the surveillance program will be in compliance with these documents with the exception that all of the RV irradiation surveillance capsules will receive a neutron flux which is at least 4 times the maximum RV neutron flux. (i.e., the lead factor for all the capsules will be at least 4)

ASTM E 185-82 recommends that the surveillance capsule lead factors (the ratio of the instantaneous neutron flux density at the specimen location to the maximum calculated neutron flux density at the inside surface of the RV wall) be in the range of one to three.

More recent versions of ASTM E 185 acknowledge that it may not be possible to position capsules in low lead factor locations due to the design of the RV internals. ASTM E 185-02 recommends that plants with lead factors greater than five should provide a method of verifying the validity of the accelerated irradiation data. This verification may be accomplished by the inclusion of a reference material.

Requested Information

1. If any surveillance capsules will have a lead factor greater than five, describe how the validity of the accelerated irradiation data will be verified.

2. For those surveillance capsules with lead factors greater than 3, justify that the surveillance specimens in the capsules will provide metallurgically meaningful data, in terms of the expected design life and/or licensed life of the RV, including possible license renewal terms, based on the fluences these capsules are projected to receive.

RAI 5.3.1-3

Background

In Amendment 97, in FSAR Section 5.2.4.2, the description of the orientation of the Charpy V-Notch specimens used to determine the initial USE of the RV beltline region was changed from "transverse" to "tangential and axial." Section 5.4.3.6 of the FSAR was also modified in Amendment 97 to change the description of the specimen orientation from "longitudinal and transverse" to "tangential and axial."

ASTM E 185-82 Section 6.2 states:

> "The tension and Charpy specimens from base metal shall be oriented so that the major axis of the specimen in parallel to the surface and normal to the principal rolling direction for plates, or normal to the major working direction for forgings as described in Section III of the ASME Code. The axis of the notch of the Charpy specimen for base metal and weld metal shall be oriented perpendicular to the surface of the material; for the HAZ specimens, the axis of the notch shall be as close to perpendicular to the surface as possible so long as the entire length of the notch is located within the HAZ."

The requirements of ASTM E 185-82 with respect to specimen orientation are consistent with the ASME Code, Section III, NB-2322.2. The Nuclear Regulatory Commission's (NRC) Branch Technical Position 5-3, "Fracture Toughness Requirements," uses the term "weak direction", defined as “transverse to the direction of maximum working.” NRC BTP 5-3 generally uses the terms “transverse” and “longitudinal” to describe the weak and strong specimen orientations.

Note (b) to Table B-1 in the PTLR (Ref. 4) indicates that the tangential direction is the strong direction and the axial direction is the weak direction. The FSAR changes appear to be for consistency with the PTLR.

Requested Information

For the WBNP-2 RV beltline materials, clarify the orientation of the Charpy specimens for the base metal in terms of the language used in ASTM E 185-82.

RAI 5.3.1-4

Background

Per 10 CFR 50.61, the pressurized thermal shock reference temperature (RT_{PTS}) for the RV beltline materials must be calculated for the end-of-license (EOL) fluence. 10 CFR 50.61 defines the *EOL fluence as* the best-estimate neutron fluence projected for a specific vessel beltline material at the clad-base-metal interface on the inside surface of the vessel at the location where the material receives the highest fluence on the expiration date of the operating license.

In order to comply with GDC 14 and 31 related to ensuring that the RCPB will behave in a nonbrittle manner and that the probability of rapidly propagating fracture is minimized, the RV materials must meet 10 CFR Part 50 Appendix G, which requires that the Upper Shelf Energy (USE) remain above 50 ft-lbs through the expiration of the plant license unless it can be demonstrated that that lower values of Charpy USE will provide margins of safety against fracture equivalent to those required by Appendix G of Section XI of the ASME Code. Regulatory Guide (RG) 1.99 provides the guidance on evaluating the drop in USE due to neutron irradiation. The projected fluence at EOL must be used to evaluate the EOL USE.

Appendixes B and C to the PTLR (Ref. 4) provide the projected USE and RT_{PTS} values based on EOL fluence values for 32 Effective Full Power Years (EFPY). However, many nuclear plants are now operating to capacity factors of 90% or more, which could make the projected EOL fluence values nonconservative if WBNP-2 achieves similar efficiency.

Requested Information

Given that nuclear power plants are now typically operating at capacity factors of 90% or greater, justify that a best estimate EOL neutron fluence based on 32 EFPY is appropriate and conservative, given that 32 EFPY is based on a capacity factor of 80% over a 40-year license. If a greater EFPY value should be postulated, provide updated Appendixes B and C to the PTLR which recalculate the USE and RT_{PTS} values for the WBNP-2 RV based on new EOL neutron fluence values.

RAI 5.3.1-5

Background

10CFR 50 Appendix G, requires the values of RT_{NDT} and Charpy USE for the RV beltline materials, including welds, plates and forgings to account for the effects of neutron radiation, including the results of the surveillance program of Appendix H of this part.

In order to determine the changes in reactor vessel fracture toughness from neutron irradiation, the RV wall fluence must be determined. NRC Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," describes acceptable approaches for fluence determinations.

FSAR Section 5.4.3.6.2 describes the methodology of the calculation of the fast neutron flux received by the surveillance samples. In FSAR Section 5.4.3.6.2.1, "Reference Forward Calculation, " in Amendment 97, the applicant included new material on the neutron transport calculation.

Requested Information

The first paragraph of FSAR Section 5.4.3.6.2.1 refers to legacy references for Unit 2 that predate the development of ENDF-B/VI, and hence are unacceptable for current calculations, according to RG 1.190 recommendations. The second paragraph, however, refers to forward transport calculations performed using more up-to-date methods for analysis of Capsule W and subsequent capsules. Please clarify the difference between these two paragraphs.

RAI 5.3.1-6

Background

With respect to special processes used to fabricate the RV (e.g. welding) SRP Section 5.3.1 essentially states that the requirements of GDC 1 and 30 and 10 CFR 50.55a regarding quality standards are met by compliance with the provisions of the ASME Code, Section III, for fabrication of components, when the appropriate code symbols are affixed and appropriate certifications made by the manufacturer or installer.

With respect to special methods for nondestructive examination, SRP Section 5.3.1 further states that the requirements of GDC 1 and 30 and 10 CFR 50.55a regarding quality standards are met by compliance with the ASME Code, Section III, for fabrication nondestructive testing, and that the acceptance criteria for examination of the RV and its appurtenances by nondestructive examination are those specified in ASME Code Section III, NB-5000.

In FSAR Section 5.4.4.2, "Penetrant Examinations," in Amendment 97 the applicant changed the description of the liquid penetrant examinations of the core support block attachment welds. The description previously stated the core support block attachment welds were inspected by dye penetrant after first layer of weld metal and after each 2 inches of weld metal. The description now states the core support block attachment welds were inspected by dye penetrant after each ½ inch of weld metal.

The code of record for the WBNP-2 RV is the ASME Code, 1971 edition through 1973 addenda. ASME Code, Section III, Sub article NB-4433 of the code of record requires that structural attachment welds be full penetration welds except for temporary attachments and minor supports. ASME Section III, Subarticle NB-5260 states that structural attachment welds to pressure-retaining material shall be examined by either the magnetic particle or liquid penetrant method. The ASME Code does not provide any requirements for the increment of liquid penetrant examination, if performed, for structural attachment welds. However, inspecting the core support block attachment welds after each ½ inch of weld is consistent with the requirements of ASME Code, Section III, NB-5245 for fillet welded and partial penetration welded joints. ASME Code, Section III, NB-5245 requires such welds to be examined progressively using either the magnetic particle or liquid penetrant methods, with the increments of examination being the lesser of one half of the maximum welded joint dimension measured parallel to the center line of the connection or ½ in. (13 mm).

Provided that the core support block attachment welds are full penetration welds, all ASME code requirements are met and the specified nondestructive examination requirements do not conflict with the ASME code requirements.

Requested Information

1. Are the core support block attachment welds full penetration welds?

2. Provide the basis for the nondestructive examination requirements for the core support block attachment welds.

RAI 5.3.2-1

Background

10 CFR Part 50 Appendix G requires that pressure-temperature (P-T) limits must be at least as conservative as limits obtained by following the methods of analysis and the margins of safety of Appendix G of Section XI of the ASME Code. The applicant provided proposed P-T limits in Reference 4. In the process of performing confirmatory calculations to check the WBNP-2 P-T curves, for the 100^{o}F/hour heatup, the allowable pressure calculated by the staff at lower temperatures is significantly less than the temperature calculated by the applicant using the methodology of Reference 6. The staff used the methodology of the ASME Code, Section XI, Appendix G to calculate the allowable pressures. The staff also observed that the thermal stress intensity factors K_{It} calculated by the applicant were generally significantly less than those calculated using the equations of the ASME Code, Section XI, Appendix G. The staff also observed that the metal temperatures given in the PTLR are higher than the corresponding metal temperatures calculated using ASME Code Section XI Appendix G Figure G-2214.2, up to a coolant temperature of 150^{o}F, above which the ASME metal temperatures are higher. The combination of the higher K_{It} and lower metal temperatures (resulting in a lower K_{Ic}) results in a lower allowable pressure being calculated using the ASME Code Appendix G methods, although the ASME values converge with the applicant's values as temperatures increase, and are within 1% of the applicant's values at 170^{o}F.

The staff used the simple equation from ASME Code Section XI, Appendix G, paragraph G-2214.3 to calculate the maximum K_{It} as a function of heatup rate:

$K_{It} = 0.753 \times 10^{-3} \times HU \times t^{2.5}$

Where:

HU is the heatup rate in $^{o}F/hr$

t = wall thickness in inches

The staff used ASME Code Section XI, Appendix G, Figure G-2214-2 to determine the temperature difference from the coolant at a specific wall depth.

The PTLR provides the above ASME Code Section XI, Appendix G equation for K_{1t} from paragraph G-2214.3 as equation (6).

ASME Code Section XI Appendix G, paragraph G-2214.3 also provides the following alternative equation for the thermal stress intensity of an outside surface defect during heatup (reproduced as Equation (8) in the PTLR:

$K_{It} = (1.043C_0 + 0.630C_1 + 0.481C_2 + 0.401C_3)^* \sqrt{\pi a}$

The coefficients C_0, C_1, C_2 and C_3 are determined from the thermal stress distribution at any specified time during the heatup or cooldown using:

$\sigma(x) = C_0 + C_1(x/a) + C_2(x/a)^2 + C_3(x/a)^3$

PTLR Section 3.2 notes that equations 3, 7, and 8 were implemented in the OPERLIM computer code, which is the program used to generate the pressure-temperature (P-T) limit curves. Section 3.2 of the PTLR further states that the P-T curve methodology is the same as that described in Section 2.6 of Reference 6 (equations 2.6.2-4 and 2.6.3-1).

However, Equation 2.6.2-4 (for the steady state analysis) and 2.6.3-1 (for the finite heatup and cooldown rate analyses) of Reference 6 provide a different method of calculating K_{It} than described by the above as equations, as described by the following equations:

$K_{1p} = 1.1M_K\sigma_p \sqrt{\pi a/Q}$ (2.6.2-4)

$Q = \Phi^2 - 0.212(\sigma_p/\sigma_y)$

$K_{1t} = [\sigma_m 1.1M_k + \sigma_b M_b]\sqrt{\pi a/Q}$ (2.6.3-1)

$Q = \Phi^2 - 0.212(\sigma_m + \sigma_b/\sigma_y)$

Where:

σ_m = constant membrane stress component from the linearized thermal hoop stress distribution,

σ_b = linear bending stress component from the linearized thermal hoop stress distribution,

M_k = correction factor for membrane stress

M_b = correction factor for bending stress, as a function of relative flaw depth (a/t)

Q = flaw shape factor modified for plastic zone size,

Φ = is the elliptical integral of the 2^{nd} kind (Φ = 1.11376 for the fixed aspect ratio of 3 of the code reference flaw), 0.212 = plastic zone size correction factor,

σ_p = pressure stress,

σ_y = yield stress,

1.1 = correction factor for surface breaking flaws,

a = crack depth of ¼ t, and

K_{Ip} = pressure stress intensity factor.

The staff notes that when the applicant's K_{It} and metal temperature values were input to the ASME Code Section XI, Appendix G equations for allowable pressure, the values obtained are identical to the applicant's allowable pressures. However, the staff requires additional information on how the applicant's K_{It} and metal temperatures were determined.

Requested Information

In order to complete our review of the P-T limits, the staff requests the following information:

1. Clarify which set of equations was used to determine the K_{It} values used as input to the P-T curve calculation.
2. If equation (8) of the PTLR was used, elaborate on how the constants C_0, C_1, C_2 and C_3 were determined, and how the thermal stress distribution σ(x) was determined.
3. If Reference 2 equations 2.6.2-4 and 2.6.3-1 were used, describe how the membrane (σ_m) and bending (σ_b) stresses were determined. Specifically, how was the initial stress profile determined prior to the linearization procedure? Also, what value was used for the yield stress?
4. With respect to the determination of the crack tip metal temperatures:
    a. Describe the boundary conditions that were assumed, particularly at the vessel outer diameter, and,
    b. Describe the methodology for calculating the temperatures.

RAI 5.3.2-2

Background

In order to comply with GDC 15 as it relates to the reactor coolant system (RCS) being designed with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operations, including anticipated operational occurrences, and 10 CFR Part 50 Appendix G with respect to fracture toughness requirements for the RV, SRP Section 5.2.2 provides guidance for the design of the low-temperature overpressure protection (LTOP) system or the cold overpressure mitigation system (COMS). The LTOP system or COMS should be in accordance with the requirements of NRC Branch Technical Position (BTP) 5-2, "Overpressurization Protection of Pressurized-Water Reactors While Operating at Low Temperatures," and that the LTOP system or COMS should be operable during startup and shutdown conditions below the enable temperature defined in paragraph II.2 of BTP 5-2.

Technical Specification B 3.4.12, "Cold Overpressure Mitigation System (COMS)," refers to the COMS arming temperature specified in the PTLR. However, the COMS arming temperature is not specified in the PTLR. Technical Specification B 3.4.12 also states that "The PTLR provides the maximum allowable actuation logic setpoints for the power operated relief valves (PORVs)." However, the PORV setpoints are not provided in the PTLR.

Requested Information

1. Provide the COMS arming temperature for WBNP-2. The PTLR must be revised to incorporate this information.

2. Provide the PORV setpoints for WBNP-2. The PTLR must be revised to incorporate this information.

RAI 6.1.1-1

Background

In order to comply with GDC 41 as it relates to control of the concentration of hydrogen in the containment atmosphere following postulated accidents to assure that containment integrity is maintained, SRP Section 6.1.1 recommends that hydrogen generation resulting from the corrosion of metals by containment sprays during a design-basis accident should be controlled as described in RG 1.7, "Control of Combustible Gas Concentrations in Containment," Regulatory Position C.6 (note, the SRP recommendation was based on Revision 2 to RG 1.7, Regulatory Position C.6 is now Regulatory Position C.4 in Revision 3 to RG 1.7).
RG 1.7, Regulatory Position C.4 states:

> Materials within the containment that would yield hydrogen gas by corrosion from the emergency cooling or containment spray solutions should be identified, and their use should be limited as much as practicable.

10 CFR 50.44 requires that all water-cooled reactor construction permits or operating licenses under Part 50 issued after October 16, 2003 comply with the following:

> All containments must have an inerted atmosphere, or must limit hydrogen concentrations in containment during and following an accident that releases an equivalent amount of hydrogen as would be generated from a 100 percent fuel clad-coolant reaction, uniformly distributed, to less than 10 percent (by volume) and maintain containment structural integrity and appropriate accident mitigating features.

FSAR Section 6.2.5 states that the combustible gas control system of the containment air return system, the hydrogen analyzer system (HAS) and the hydrogen mitigation system (HMS) conform to 10 CFR 50.44 requirements. FSAR Section 6.2.5.1 further states that:

> In an accident more severe than the design-basis loss-of-coolant accident (LOCA), combustible gas is predominantly generated within containment as a result of the following:
>
> (1) Fuel clad-coolant reaction between the fuel cladding and the reactor coolant.

(2) Molten core-concrete interaction in a severe core melt sequence with a failed reactor vessel.

It appears that corrosion of materials has been removed from the design bases of the HMS in FSAR Section 6.2.5.

Additionally, FSAR Section 6.2.1.3.3 includes the following as an input assumption for the containment pressure analysis:

> Hydrogen gas was added to the containment in the amount of 25,230.2 Standard Cubic Feet (SCF) over 24 hours. Sources accounted for were radiolysis in the core and sump post-LOCA, corrosion of plant materials (aluminum, zinc, and painted surfaces found in containment), reaction of 1% of the Zirconium fuel rod cladding in the core, and hydrogen gas assumed to be dissolved in the reactor coolant system water. (This bounds tritium producing core designs.)

If the potential for generation of hydrogen gas due to corrosion of reactive metals is insignificant compared to the generation of hydrogen due to the fuel clad-coolant reaction and the molten core-concrete interaction in a severe core melt sequence with a failed RV, it may be unnecessary to limit or quantify reactive metals in order to comply with 10 CFR 50.44.

However, SRP Section 6.1.1 still recommends that hydrogen generation due to corrosion of reactive metals be addressed. The information in FSAR Section 6.2.1.3.3 also implies a design basis assumption on the amount of aluminum, zinc and coatings containing reactive metals.

<u>Requested Information</u>

1. Was the potential for generation of hydrogen due to corrosion of reactive metals such as zinc or aluminum considered in the design of the combustible gas control system, or other analyzes such as containment pressure? If so, describe how this contribution was evaluated.

2. If the contribution of hydrogen from reactive metals was not evaluated, justify why this contribution was not evaluated.

3. If the contribution of hydrogen from reactive metals was evaluated, discuss the measures taken to ensure that the use of materials that could yield hydrogen gas by corrosion from the emergency cooling or containment spray solutions is limited as much as practicable, and is maintained within design basis limits.

<u>References</u>

1. Overview Report on Zinc Addition in Pressurized Water Reactors—2004, 1009568 Final Report, December 2004, Electric Power Research Institute

2. Pressurized Water Reactor Primary Water Zinc Application Guidelines 1013420 Final Report, December 2006, Electric Power Research Institute

3. PWR Axial Offset Anomaly (AOA) Guidelines, Revision ,1008102, Final Report, June 2004, Electric Power Research Institute

4. WCAP-17035-NP, “Watts Bar Unit 2 Heatup and Cooldown Limit Curves for Normal Operation and PTLR Support Documentation,” Revision 2, December 2009 (ADAMS Accession No. ML100550651)

5. ASTM E-185-82, “Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels”

6. WCAP-14040-A, “Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves,” Revision 4, May 2004 (ADAMS Accession No. ML0501202094)

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From: Wiebe, Joel

Created By: Joel.Wiebe@nrc.gov

Recipients:

WDCrouch@TVA.gov ('Bill Crouch')

Tracking Status: None

GArent@tva.gov ('Gordon Arent')

Tracking Status: None

DTLangley@TVA.gov ('Tony Langley')
Tracking Status: None
WBN2HearingFile.Resource@nrc.gov (WBN2HearingFile Resource)
Tracking Status: None

Post Office:
HQCLSTR02.nrc.gov

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