NRC FORM 757
NRC MD 10.158
(3-2007)

U.S. NUCLEAR REGULATORY COMMISSION

NON-CONCURRENCE PROCESS

SECTION A - TO BE COMPLETED BY NON-CONCURRING INDIVIDUAL

TITLE OF DOCUMENT
HUT issues' inputs to Braidwood & Byron inspection reports 08-03.

ADAMS ACCESSION NO.

DOCUMENT SPONSOR
Richard Skokowski

SPONSOR PHONE NO.
630-829-9620

NAME OF NON-CONCURRING INDIVIDUAL
Gerard O'Dwyer

PHONE NO.
630-829-9624

[x] DOCUMENT AUTHOR [ ] DOCUMENT CONTRIBUTOR [ ] DOCUMENT REVIEWER [x] ON CONCURRENCE

TITLE
Reactor Inspector

ORGANIZATION
Region III, DRS, EB2

REASONS FOR NON-CONCURRENCE

This is as much as my writeup of some of my reasons as I had time to document for my nonconcurrence to the HUT inputs to Braidwood & Byron reports 08-03. GFO 7/28/08

The inputs to the Byron and Braidwood reports numbered 08-03 that I was required to write GO 7/28/08 [redacted] did not ~~did not~~ state that there still exists an immediate safety concern. The inputs also state that the performance deficiency (of failing to recognize that the HUT will catastrophically fail and not maintaining minimum HUT levels to prevent such failure) is only a green NCV with only a very low safety significance. The input also states that the performance deficiency (of failing to evaluate the potential for water hammers) is only a green NCV with only a very low safety significance. The inputs also state that failure to correctly analyze HUT failures is only a minor violation.

I respectfully disagree with the inputs because they are very inadequate to document my concerns and to notify the licensee and the public officially of my concerns. [redacted] believe that after a LOCA the offsite consequences will be worse that analyzed for a LOCA (see section 1). [redacted]

I respectfully disagree with the inputs because I believe that the NCV for failing to evaluate for water hammers can not be determined to be only a Green NCV because there was and still is an immediate safety concern. [redacted] stated that there was no immediate safety concern. The input however supports my contention that the there is an immediate safety concern because the input even states that no water hammer analysis has

[x] CONTINUED IN SECTION D

SIGNATURE
Gerard O'Dwyer Gerard O'Dwyer 7/29/08

DATE
7/28/08

SUBMIT FORM TO DOCUMENT SPONSOR AND COPY TO YOUR IMMEDIATE SUPERVISOR

NON-CONCURRENCE PROCESS

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been done so it is unknown at this time if the RH relief valve to HUT piping will catastrophically fail during a discharge. The input states: "The inspectors also noted that although existing seismic supports would provide margin to accommodate transient loads in the direction of the support restraint, the postulated dynamic transient **might** create piping forces in directions that **were not** restrained by the existing pipe supports." The input resolves this by stating that the licensee entered the requirement to do water hammer analyses in the future into the corrective action program (CAP) and "The licensee also implemented compensatory actions, including raising the water level in the HUT to above the discharge line connection at the HUT nozzle in order to **ensure that quenching would occur**." I believe that the minimum 40% level (which can correspond to as little as 4.1 inches of quenching water) has a very high probability of not being adequate (see section 0 below and the URI concerns in the inputs). There was and still is no basis for the NRC to assume that 40% minimum level is adequate. In fact, by a preponderance of the evidence significantly more than 40% minimum level is required.

I respectfully disagree with the inputs because I believe there was and still is an immediate safety concern for the performance deficiency of not maintaining minimum HUT levels (see Section 1 & 2) however the input does not state that there is an [redacted] stated that there was no immediate safety concern. I disagree because there is a high probability that the performance deficiency of not maintaining minimum HUT levels has a higher safety significance than green because there are other performance deficiencies that I have not even been allowed to document adequately (see Section 1 & 2). There are also other probable performance deficiencies that I have not even been allowed to document questions about in the URI (See sections 4 &5). I believe the report must ensure that the potential safety significance of these performance deficiencies be carefully determined [redacted] I believe it is also necessary to not rush the inspection and documentation of these concerns because of the complexity of these issues and the disingenuousness and uncooperativeness of the licensee. [redacted] The evaluation was conducted independently by the Byron Design Engineering Department. During the course of the evaluation, input and review was obtained from Braidwood Design Engineering, Sargent & Lundy, Byron and Braidwood Plant Engineering, and EGC Corporate Engineering. In addition, Westinghouse was contracted to assist in evaluating the concerns." Teams of licensee personnel and contractors have been trying to minimize these concerns for months [redacted]

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[redacted] "The evaluation was conducted independently by the Byron Design Engineering Department. The independent evaluator has been in the nuclear power industry for 27 years, and has worked in the Byron Design Engineering Department for 14 years." The evaluator from the Byron "Design" Engineering Department was not independent of the 2 NCVs and minor violation because they were all design performance deficiencies and violations of 10 CFR 50, Appendix B, Criterion III Design Control. The NRC should not accept the licensee's responses but should demand more information. At the very least the NRC should fully inspect these issues because of the lack of cooperation by the licensee. At the very least the NRC should fully and officially and formally document all followup issues in the URI.

I respectfully disagree with the inputs that the failure to correctly analyze HUT failures is only a minor violation (see section 5).

I respectfully disagree with the inputs that the intersystem LOCA potential of the CVCS, SI and RH system relief valves lifting has been adequately inspected and I believe that it was and is still an immediate safety concern. (see SECTION 6)

The deficiencies, questions and issues in the URI itself document that the true risk significance of the performance deficiencies cannot at this time be determined to be only green and not greater than green and that there still is an immediate safety concern.

While reviewing the PORVs for these concerns I identified an issue that appears to be a Tech Spec Violation with a potential risk and safety significance of at least Green and possibly higher. See Section 7.

The Byron Unit 2 RH suction relief valves discharged on May 1, 2007 as a result of RCS water being driven through all four of the CLOSED RH suction isolation valves 2RH8701A/B and 2RH8702A/B when the RCS pressure was recorded to be 385 psig. These isolation valves are supposed to protect the RCS from having an ISLOCA through the RH system even if the RCS pressure is 2350 psig. This appears to be a very significant performance deficiency for these valves to have allowed RCS water through that lifted the RH suction relief valves set at 450 psig. I believe this needs to be at least a followup concern documented in the URI.

SECTION 0: There is a high probability that the performance deficiency has a higher safety significance than green and still is an immediate safety concern because the licensee's

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presently-required minimum HUT level of 40% level is probably inadequate. I was not allowed to document the following issue. Additional verifier comment #1 of Westinghouse calculation CN-CRA-08-9 stated the 40% level provided a depth of water over the top of the inlet piping into the HUT that **could be as little as only 4.1 inches of water** due to instrument uncertainty. The calculation, however, assumes the depth of water over the top of the inlet piping into the HUT could be as little as 4.25 inches of water due to instrument uncertainty. The calculation's author acknowledged Additional verifier comment #1 as correct but did not correct the minimum case because he said it was close enough. Either value indicates that the 40% level is probably inadequate to ensure adequate quenching.

SECTION 1: There is a high probability that the performance deficiency has a higher safety significance than green and is an immediate safety concern because the minimum HUT level controls are only required in Mode 4. The licensee stated that before a refueling outage (MODEs 1-3) they attempt to operate the HUT with a level as close as possible to the previous minimum level of 5%. The Byron controls require a minimum of 40% only when the plant is actually in Mode 4 with the RH relief valves lined up to the HUT. The Braidwood controls are probably the same but the licensee did not send enough information to address my concern.

One example of an undocumented aspect that indicates a high probability that the performance deficiency has a higher safety significance than green is the following paragraph that was in my input submission but was removed by Mrs. Stone:

"There is also an immediate safety concern whenever any of the four plants are in MODEs 1 through 4. Minimum HUT level controls are required in MODEs 1 through 4 because there exist very high probabilities that during switchover to containment sump recirculation as required after almost all sizes of LOCAs the RHR pumps will lift the CVCS and SI pumps' suction relief valves [which also relieve to the HUTs] which are set at about 220 psig and 200 psig respectively [because as stated in UFSAR section 6.3.2.8, "Manual Actions" and UFSAR table 6.3-7, "Sequence of Switchover Operations", the CVCS and SI pumps are piggybacked on the RHR pumps after a LOCA and] because the shutoff head of the RHR pumps are about 190 psig and the containment sump pressure will be about 50 psig so the RH pumps could easily cause a pressure transient up to 240 psig which would lift all the [CVCS and SI suction] reliefs. The RH pumps also could easily cause a pressure transient up to 200 psig which would lift the CVCS reliefs which also discharge to the HUT. According to the uprate analysis the sump water would be as high as 259 degrees Fahrenheit [corresponds to 20 psig]. Switchover to Containment Sump from RWST time period has a high probability of hydraulic pressure transients. Also during switchover there is a high probability of pressure transients (aka water hammers, hydraulic transients, etc.) even higher than 240 psig. Lifting the reliefs would also violate the Alternate Source Term (AST) required post-LOCA ECCS leakage limits [because the licensee's

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application for the AST license change and the NRC's SER (section 3.2.3) approving the AST license change had no consideration of the possibility of the CVCS, SI or RH system relief valves lifting and sending highly contaminated post-LOCA reactor coolant to the HUT]."

My writeup was removed and replaced with the vague and inadequate explanation: "The licensee had not completed their corrective actions with respect to analysis of the impact of other valves releasing to the HUT including the CVCS & SI system suction relief valves when in the piggyback mode on the RH system after a LOCA."

The writeup is very inadequate to document my concern and to notify the licensee and the public officially of my concern. [redacted]

SECTION 2: Plant needs to be able to shutdown at any time in MODEs 1 through 4 and 40% minimum level is only maintained in MODE 4.
One example of an undocumented aspect that indicates a high probability that the performance deficiency has a higher safety significance than green is the following paragraph.

There is also an immediate safety concern whenever any of the four plants are in MODEs 1 through 3 because many Tech Spec LCOs require any of the four Units to be in Hot Shutdown within 6 hours. After a plant trip any of the units could be rapidly in MODE 4 with a level in the HUT even lower than 40% and not enough time to restore the HUT level to even 40% level. The Byron controls require a minimum of 40% only when the plant is actually in Mode 4 with the RH relief valves lined up to the HUT. The Braidwood controls are probably the same. ✓

SECTION 3: There is a high probability that an immediate safety concern still exists because the minimum HUT level controls are inadequate due to other probable performance deficiencies that I have not even been allowed to document questions about in the URI. I have not been allowed to adequately document that Calculation CN-CRA-08-9, Revision 0, "Byron/Braidwood RHUT Response to Opening of the RHR Relief Valve", February 14, 2008 uses an excessive

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pressure for the failure pressure of the top of the HUT. I have not been allowed to document that the calculation ignores the design temperature limit of 200 F for the HUT which is stated in UFSAR section 15.7.2.1. The HUT is designed for only 200 F even though design pressure is 15 psig (29.7 psia). The temperature of 15 psig (30 psia) steam is about 245 F. At 15 psig the HUT is about 45 F past design. Even at 1 psig the HUT is about 15 F past design. Thermal stresses are very important on tanks and can cause failures and there is no analysis that indicates any temperature over 200 F is acceptable for the HUT.

SECTION 4: I do not believe that enough inspection has been done to conclude [redacted] that the HUT relief valve discharging steam and radioactivity to the HUT room is not a personnel hazard and radioactive contamination hazard.

SECTION 5: I do not have time to document this concern.

SECTION 6: I do not have time to document this concern.

Section 7: I do not have time to document this concern.

Section 8: I did not have ~~time~~ or assistance to document the ASME Code violation of routing the RH suction and discharge relief valves which are designed to release 350°F and 450 psig steam to the HUTs which are designed for maximum temperature of 200°F and 15 psig.

1/29/08

I have many more ~~concerns~~ objections to the inputs that I have not had time to document.

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NON-CONCURRENCE PROCESS

TITLE OF DOCUMENT
HUT issues' inputs to Braidwood & Byron inspection reports 08-03.

ADAMS ACCESSION NO.

SECTION B - TO BE COMPLETED BY NON-CONCURRING INDIVIDUAL'S SUPERVISOR

NAME
Ann Marie Stone

TITLE
Chief, Engineering Branch 2

PHONE NO.
630-829-9729

ORGANIZATION
DRS, R3

COMMENTS FOR THE DOCUMENT SPONSOR TO CONSIDER

☐ I HAVE NO COMMENTS

☑ I HAVE THE FOLLOWING COMMENTS

See Attached

☐ CONTINUED IN SECTION D

SIGNATURE
Ann Marie Stone

DATE
11/12/08

SUBMIT THIS PAGE TO DOCUMENT SPONSOR

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(3-2007)

U.S. NUCLEAR REGULATORY COMMISSIC

NON-CONCURRENCE PROCESS

TITLE OF DOCUMENT

HUT issues' inputs to Braidwood & Byron inspection reports 08-03.

ADAMS ACCESSION NO.

SECTION C - TO BE COMPLETED BY DOCUMENT SPONSOR

NAME

AnnMarie Stone

TITLE

Chief, Engineering Branch

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630-829-9729

ORGANIZATION

DRS, R3

ACTIONS TAKEN TO ADDRESS NON-CONCURRENCE

See Attached

☐ CONTINUED IN SECTION D

NON-CONCURRING INDIVIDUAL *(To be completed by document sponsor)*:

☐ CONCURS

☐ NON-CONCURS

☐ WITHDRAWS NON-CONCURRENCE

Non-concurred on draft IR 2008003 input for Byron and concurred on Byron IR 2008009 following completion of addition inspection. 11/12/08

IR 2008009 may be found in ADAMS at ML083180104.

Supervisor/Document Signer/Document Owner Response

Mr. O'Dwyer requested that this non-concurrence be classified as a public document and acknowledged that some material would need to be redacted. As appropriate, information not pertaining to the inspection report input has been redacted; therefore, resolution of those insights will not be addressed in my response.

Because I am Mr. O'Dwyer's supervisor, the Document Sponsor, and the Document Signer, the Deputy Division Director, Anne Boland, will serve the function of the Document Signer, in that, she will have the responsibility for the final decision on the contents of the inspection report input. However, I will retain the responsibility of signing the inspection report input memorandum to the Byron and Braidwood Senior Resident Inspectors.

I reviewed Mr. O'Dwyer's non-concurrence report and concluded that his concerns could be summarized into five areas:

1. Immediate Safety Concern
2. Outstanding Technical Issue
3. Significance of Findings
4. Documentation of Unresolved Item
5. Other Issues

I have addressed each of the concerns below:

1. Immediate Safety Concern

    In the non-concurrence report, Mr. O'Dwyer stated several times that even though the licensee has raised HUT level to at least 40% when in Mode 4 operation, there still exists an immediate safety concern. He states that after a LOCA, the offsite consequences will be worse than analyzed. He also stated that the NCV for failing to evaluate for water hammers could not be determined to be only Green because there was a still is an immediate safety concern. Mr. O'Dwyer stated that a safety concern exists whenever any of the plants are in Modes 1-3 because many LCOs require any of the four Units to be in Mode 4 within 6 hours. He further states that after a plant trip, the units could be rapidly in Mode 4 with the level in the HUT lower than 40%. The Byron controls require a minimum of 40% level when the plant is in Mode 4 with the RH relief valves lined up to the HUT.

    In my assessment of the immediate safety concern, I considered two concepts: (1) risk significance as it is defined by the ROP and (2) potential scenarios.

    With respect to the risk significance, the failure of the HUT does not impact the likelihood of an initiating event and will not result in an increase in core damage frequency. The failure of the HUT does not impact Large Early Release Frequency because the LERF is dependent on failures of containment resulting in large releases capable of exceeding Part 100 limits. Failure of the HUT could be considered an intersystem LOCA (as discussed in IN 90-05); however, additional component failures or operator errors would need to occur to result in releases equivalent to a large break LOCA with fuel damage. Therefore, based on PRA risk insights, failure of the HUT does not constitute an immediate safety issue.

With respect to potential scenarios, I considered normal shutdown situations as well as accident/transient conditions. For normal shutdown situations, the RH suction relief valves are isolated from the reactor piping through two normally closed MOVs in Modes 1 – 3. Leakage past the two discharge check valves could increase the pressure of the RH piping and on at least two occasions, caused the relief valves to lift. The amount of energy in these lifts would be small because it presented leakage past closed seats. As the differential pressure across the check valves increased (startup), the leakage stopped (and was verified by the licensee). In Mode 4, the suction relief valves would be exposed to reactor pressure when the operators open the RHR suction valves for residual heat removal operation. At this point, I considered the development of the issue itself. The UFSAR is clear that damage to the HUT was not expected to occur during the lifetime of the plant. The licensee has shown that insufficient level when an RHR suction relief valve lifts could result in damage to the HUT. In addition, as discussed in Generic Letter 90-06, the NRC recognized the importance of protecting against low temperature overpressure events and required licensees to take actions to prevent these events. The actions included placing pumps in pull-to-lock to prevent inadvertent starts and other administrative actions. I recognize that the RHR relief valves would lift prior to the PORVs during an LTOP transient; however, I also acknowledge that these administrative actions also greatly reduced the potential onset of an LTOP transient. The USFAR also provides an analysis for two significant failure modes of the HUT: a leak resulting in liquid release and a fault resulting in gaseous release. Both of these events are well below Part 100 limits. In addition, the licensee now requires HUT level to be 40% when in Mode 4.

With respect to emergency operations, when RHR is used for inventory control (injection mode), suction is taken from the RWST or from the containment sump. At some point in recovery, suction may be swapped to the cold leg or hot leg for heat removal. In all of these scenarios, suction pressure would be well below the lift setpoints of the relief valves.

With respect to the offsite consequence of a LOCA being higher than expected, the circumstances (i.e. both CVCS and SI suction relief valves opening) will be examined as part of the URI. As Mr. O'Dwyer indicated, the RHR pump shutoff is about 200 psid and the suction relief valves for the charging and safety injection pumps are set at about 220 psig. Expected pressure drops (piping, elbows, valves, etc.) should result in suction pressures much lower than the SI/CVCS relief valve settings. (Aside: If the relief valves are set too low, this would be followed up as another performance deficiency and is separate from the HUT issue.) In addition, the relief valves pass only 25 gpm – significantly less than the RHR suction relief valves. Although I believe this should be looked into, I don't believe it poses a problem with respect to HUT rupture.

In addition, during component design basis inspections, the inspectors verify the design and operation safety related pumps. During Braidwood CDBI (Inspection report 05000456/2007009; 05000457/2007009), the inspectors reviewed the safety injection and residual heat removal. I am not certain that this element was inspected; therefore, it is appropriate to verify this design feature during a followup inspection. If this issue is not resolved by February 2009, the Byron CDBI team will followup on the adequacy of the CVCS/SI suction relief valves.

In conclusion, I do not agree that an immediate safety concern exists.

2. Outstanding Technical Issue

Mr.O'Dwyer states that the minimum 40% level (which can correspond to as little as 4.1 inches of quenching water) has a very high probability of not being adequate. Mr. O'Dwyer further stated that by a preponderance of the evidence significantly more than 40% minimum level is required. He also stated that the intersystem LOCA potential of the CVCS, SI, and RH system relief valves lifting has not been adequately inspected.

I agree with Mr. O'Dwyer that additional inspection is warranted. This was the basis for the planned followup inspection originally scheduled for December 2007. It continued to be the plan as discussed in February 2008 with the creation of an unresolved item and with the writeup in question now. If these corrective actions are shown to be inadequate, additional enforcement actions (in accordance with IMC 0612, IMC 0609, and the Enforcement Policy) will be taken.

3. Significance of the Issues:

In the non-concurrence report, Mr. O'Dwyer states that the violations should be more significant because of the potential to catastrophically fail the HUT.

As with all findings, we used IMC 0612 and IMC 0609 to determine the significance of the three violations. The first step described in IMC 0612 is to determine the performance deficiency. Mr. O'Dwyer had no concerns with this step. At this point, each finding will be presented separately.

a) NCV for failure to maintain level:. Mr. O'Dwyer believes this issue should not be Green.

   To continue the discussion above, the second step is to determine whether traditional enforcement is applicable. The issue of potential damage to the HUT if other events or failures occurred (operator error during cooldown/heatup, relief valve failure to seat, inadvertent opening of the RHR suction valves, etc.) does not constitute an issue with "actual or potential safety consequence." IMC 0612 defines actual consequence as overexposure, actual radiation release greater than 10 CFR Part 20 limits, for example.

   The next step involves determination of minor significance. We determined the failure to maintain HUT level was more than minor because it was associated with a cornerstone attribute and had an impact on the cornerstone objective. The next step in the process involves determining significance based using the SDP (IMC 0609). In Attachment 0609.04, it states, "do not include hypothetical conditions (single failure criteria) or speculate on the "worst case" potential degradations as an input to the official SDP result. However, a bounding determination of significance may be made by assuming a worst case condition.

For example, assume a complete loss of function even if unsupported by the facts known at that time."

In Modes 1-3, RHR suction relief valve(s) are isolated from the RCS. The relief valves will not open unless we assume additional valve failures (the closed RHR suction valves open). As stated above, speculations of other equipment failures is not allowed.

In Modes 4-5, we assumed the RHR suction relief valve(s) open as designed in response to rising pressure in the RHR line. The RHR suction valves would be open; therefore, we do not need to assume a failure associated with these valves. Since the issue deals with the licensee's inability to control the HUT level, it is appropriate to assume the level was low, the relief valves are open, and the HUT is damaged. This is consistent with IMC 0609.04.

From Table 2, we determined the affected cornerstone was Barrier Integrity.

From Table 3a, we determined the following:

- Emergency Preparedness – does not apply
- Occupational Radiation Safety – does not apply. We reviewed Appendix C to confirm.
- Public Radiation Safety – does not apply. We reviewed Appendix D to confirm. The effluent release program described in this appendix relates to actual programs, not to unplanned releases resulting from a damaged HUT.

From Table 3b: We reviewed the issue with an SRA and initially, we did not answer "'yes" to any question and continued with Table 4b. The inspection report input will be revised to describe the signficiance of the issues using Appendix G as outlined below: On re-analysis, I answered "yes" to question 2 (safety of the plant in shutdown operations):

- Appendix G: Assuming Mode 4 or 5 operation, LTOP transient such that the relief valves open to the HUT, HUT damaged:
    - Checklist 1: Not applicable. The RHR relief valves would not be exposed to RCS pressure in Hot Shutdown.
    - Checklist 2: Core Heat Removal, Inventory Control, Power Availability, Reactivity Guidelines do not apply. A damaged HUT would bypass containment temporarily; however, it is isolable through closure of the RHR suction valves. We did not analyze a stuck open relief valve as this would be another failure (already assuming operator error causing increasing pressure in the RHR pipe.) This does not require a Phase 2 because it does not degrade the licensee's ability (1) to terminate a leak path, (2) to recover DHR or (3) to keep the containment intact following a severe accident. In addition, damage to the HUT did not impact the ability of the relief valves or PORVs from performing their LTOP function.
    - Checklist 3: Not applicable. With RCS open, pressure is well below the lift setpoint of the relief valves. With the RCS closed and with no inventory in the Pressurizer, it highly unlikely the RHR piping pressure would reach the relief setpoints due to the large bubble in the pressurizer.

- Checklist 4: Not applicable. Event can not occur in refueling mode of operation.

Therefore, the finding screened as Green.

b) NCV for failure to evaluate water hammers: Mr. O'Dwyer statements imply that because no water hammer analysis was completed, it is unknown if the RHR relief valve to HUT piping would catastrophically fail. He further states that the lack of analysis is resolved because the licensee initiated corrective actions (including raising level in the HUT), Mr. O'Dwyer disagrees with this resolution. Mr. O'Dwyer also stated that he did not have time or assistance to document the ASME Code violation.

This part of the inspection was conducted by a senior inspector with extensive experience in structures and supports. During the initial inspection in September and October 2007, this inspector engaged the licensee in discussion and concluded that although no analysis existed, the issue was Green.

We processed the finding through IMC 0612. Again, Mr. O'Dwyer did not have a concern with the performance deficiency. The next step is to determine whether traditional enforcement is applicable. The issue of potential damage to the HUT if other events or failures occurred (operator error during cooldown/heatup, relief valve failure to seat, inadvertent opening of the RHR suction valves, etc.) does not constitute an issue with "actual or potential safety consequence." IMC 0612 defines actual consequence as overexposure, actual radiation release greater than 10 CFR Part 20 limits, for example.

The next step involves determination of minor significance. For the failure to evaluate for water hammer, we determined the finding was more than minor because it was associated with a cornerstone attribute and had an impact on the cornerstone objective. The next step in the process involves determining significance based using the SDP (IMC 0609). We used the same logic described for the failing to maintain HUT level to determine the significance of this issue. It should be noted that when evaluating the impact of a water hammer we assumed failure of the RH piping attached to the HUT which has a lesser impact than failure of the HUT itself.

c) Minor violation for the failure to correctly analyze HUT failures. Mr. O'Dwyer disagrees that this issue is minor.

This issue was followed up by two radiation protection inspectors in late Fall 2007. These individuals reviewed the licensee's response to several questions, applicable UFSAR sections, and applicable calculations. These inspectors determined that the UFSAR analysis was incorrect as it did not reflect actual operation in the plant. However, these inspectors determined that the performance deficiency was minor based on several conservative assumptions in the calculation and because of the large margin (well under Part 100 values). This conclusion was discussed with the Plant Support Team Lead.

With respect to processing this issue using IMC 0612, we determined that traditional enforcement was not applicable because the failure to correctly analyze HUT failures does not constitute an issue with "actual or potential safety consequence,“ was not determined to be willful by the Office of Investigation, and did not impede the NRC’s ability to perform our function. We determined that this performance deficiency was similar in concept to examples 3i, 3j and 3k of Appendix E, in that, the analysis in the UFSAR used incorrect assumptions; however, the overall margin compensated for the errors. It should be noted that this is a violation that the licensee is required to address – regardless of the “minor” classification.

4. Documentation of the Unresolved Item

In the non-concurrence report, Mr. O’Dwyer states he disagreed with my direction to informally track his questions and that he was not allowed to formally document all of his concerns. For example, Mr. O’Dwyer stated that his concern related to the offsite consequences following a LOCA was not included in the inspection report input. Mr. O’Dwyer also states that there are also other probable performance deficiencies that he was not allowed to document as questions in the URI writeup. Mr. O’Dwyer stated that the NRC should at the very least fully and officially document all followup issues in the URI. He further states that the deficiencies, questions, and issues in the URI itself document that the true risk significance of the performance deficiencies cannot at this time be determined to be only Green.

This circumstances which led to the unresolved item are unusual. In most cases, URIs result when the inspector and the licensee are at a temporary stopping point – either more information is needed from the licensee or the NRC needs to perform an action such as discuss with NRR. With respect to this URI, an inspection to review the licensee’s corrective actions was planned for December 2007. Due to other circumstances, this inspection did not occur. For the next 2 months, the licensee provided copies of their corrective action program documents; however, a formal followup inspection - including interface with the licensee – did not occur. Many of the issues raised by Mr. O’Dwyer have not been discussed with the licensee; therefore, at this time, are merely questions. For example, the URI bullet regarding the Part 21 notification is a question – we don’t know if the licensee evaluated the issue against Part 21– the bullet does not reflect a concern that the licensee should have or failed to evaluate for Part 21 implications.

Because the licensee has not been engaged, I believe it is appropriate to briefly state the actual concerns and track the questions needed to follow-up these concerns. I believe that the URI input adequately documents our initial assessment of the licensee’s corrective actions and the need for further inspection. It should be noted that the philosophy on the amount of detail and length of this URI is consistent with other URIs.

5. Other Issues

- Mr. O’Dwyer states that while reviewing the PORVs, he identified an issue that appears to be a Technical Specification violation with potential risk and safety significance.

This issue was not brought to my attention prior to the writing of this non-concurrence. Therefore, I was not given an opportunity to determine whether documentation was appropriate.

I asked Mr. O'Dwyer to provide additional information. In an email dated August 11, 2008, he stated that T.S. 3.4.1, LCO 3.4.3 required cooldown rates be maintained within the limits specified in the PTLR. Figure 3.1 of the PTLR indicated that at about 355 degrees Fahrenheit, the PORV shall have a maximum setting of about 1500 psig. However, USAR Table 5.4-10 indicated that the PORV setpoint is actually 2335 psig when the RCS is at 355 degrees Fahrenheit.

I reviewed USAR Table 5.4-10 and noted an asterick which stated the PORV setpoints have remote manual control. USAR sections 5.2.2.11.1 and 7.6.9 describe the LTOP logic and stated the setpoints for the PORVs are varied based on RCS pressure and temperature. I reviewed the plant cooldown procedure and noted that the operators verify LTOP operability by performing surveillances which lower the setpoints accordingly.

Mr. O'Dwyer has not provided additional information nor communicated whether the above addressed his question.

- In the non-concurrence report, Mr. O'Dwyer includes a discussion of the Byron Unit 2 RH suction relief valves discharge event which occurred in May 2007. He concludes that this event needs to be at least a followup concern documented in the URI.

  This issue was included in the URI. I don't understand why Mr. O'Dwyer included this comment in the non-concurrence report.

- Mr. O'Dwyer also stated that he was not allowed to document his concern with a comment made by a verifier of the Westinghouse calculation. Specifically, that the HUT could be as little as 4.1 inches of water due to instrument uncertainty.

  With respect to the specific comment regarding instrument uncertainty, Mr. O'Dwyer had not included this discussion in the draft copies of inputs. However, this is an example of a question to be presented to the licensee.

- Mr. O'Dwyer commented that he was not allowed to document a concern that a minimum level should be established for all modes of operation because there exists a high probability that during switchover to containment sump recirculation, the RHR pumps will lift the CVCS and SI pumps' suction relief valves.

  This discussion was removed with the intention that Mr. O'Dwyer would more fully develop the issue with the licensee. The writeup was reduced to a brief statement questioning the licensee's evaluation of other relief valve inputs into the HUT. As stated above, during the follow up associated with the URI, we will verify that the design calculations or system lineups account for this higher pressure.

- Mr. O'Dwyer stated that he as not allowed to adequately document that Calculation CN-CRA-08-9 used an excessive pressure to fail the top of the HUT.

  This discussion was included in the inspection report input. I don't understand why Mr. O'Dwyer include this comment in the non-concurrence report.

- Mr. O'Dwyer states that not enough inspection has been done to conclude that the HUT relief valve discharging steam and radioactivity to the HUT room is not a personnel hazard and radioactive contamination hazard.

  Personnel safety hazards are addressed through OSHA. I will discuss this aspect with Mr. O'Dwyer. Contamination hazards may occur with any breach of potentially contaminated fluid lines.

Mr. O'Dwyer also stated that he did not have enough time to document his concerns in this non-concurrence. Because of his involvement in this issue over the past year, I reasoned he was extremely knowledgeable of the issues therefore, it was appropriate to authorize about 8 hours to document his objections. When Mr. O'Dwyer requested additional time, the additional time was granted. Mr. O'Dwyer was provided ample time to document his concerns.

Ann Marie Stone

Analysis: The inspectors determined that the failure to fully analyze the consequence of the lifting of RH suction relief valves which were expected occurrences was a performance deficiency warranting a significance evaluation. The inspectors determined the finding was more than minor in accordance with IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Disposition Screening," because the finding was associated with the Barrier Integrity cornerstone and maintaining the Radiological Barrier Functionality of the Auxiliary Building of a PWR. The finding was associated with the design control and procedure quality attributes of the cornerstone. Specifically, the licensee did not appropriately ensure that design analyses bounded potential RH relief valve discharges greater than 212ºF to the HUTs. Also, design requirements were not translated into licensee procedures to appropriately ensure a minimum HUT water level to ensure adequate quenching to prevent potential catastrophic HUT failures due to over-pressurization.

The inspectors evaluated the finding using IMC 0609, Attachment 0609.04, "Phase 1 – Initial Screening and Characterizations of Findings." The inspectors concluded the finding affected the safety of the reactor during refueling outages, forced outages, and maintenance outages; therefore, Appendix G, "Shutdown Operations", Checklist 2, "PWR Cold Shutdown Operation: RCS Closed and SGs Available for DHR Removal," was applicable. The inspectors determined that a phase 2 or phase 3 analysis was not required because the finding did not result in non-compliance with LTOP, did not increase the likelihood of a loss of decay removal, did not degrade the ability to terminate a leak path (relief valve and suction valves were functioning) and did not degrade the ability of containment to remain intact following a severe accident. Specifically, the concern with lifting RHR relief valves causing damage to the HUT would bypass containment temporarily; however, it is isolable through closure of the RHR suction valves or closure of the relief valves once pressure was relieved. In addition, damage to the HUT did not impact the ability of the relief valves or PORVs from performing their LTOP function. Therefore, the inspectors determined the finding was of very low safety significance (Green).

The inspectors concluded this finding did not have a cross-cutting aspect .

Analysis: The inspectors determined that the failure to evaluate the potential for water hammers was a performance deficiency warranting a significance evaluation. The inspectors determined the finding was more than minor in accordance with IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Disposition Screening," because the finding was associated with the Barrier Integrity cornerstone and maintaining the Radiological Barrier Functionality of the Auxiliary Building of a PWR. The finding was associated with the design control and procedure quality attributes of the cornerstone. Specifically, the licensee did not appropriately evaluate potential water hammer from the relief valves discharging to the HUT per design.

The inspectors evaluated the finding using IMC 0609, Attachment 0609.04, "Phase 1 – Initial Screening and Characterizations of Findings." The inspectors concluded the finding affected the safety of the reactor during refueling outages, forced outages, and maintenance outages; therefore, Appendix G, "Shutdown Operations", Checklist 2, "PWR Cold Shutdown Operation: RCS Closed and SGs Available for DHR Removal," was applicable. The inspectors determined that a phase 2 or phase 3 analysis was not required because the finding did not result in non-compliance with LTOP, did not increase the likelihood of a loss of decay removal, did not degrade the ability to terminate a leak path (relief valve and suction valves were functioning) and did not degrade the ability of containment to remain intact following a severe accident. Specifically, the concern with lifting RHR relief valves resulting in waterhammer damage would bypass containment temporarily; however, it is isolable through closure of the RHR suction valves or closure of the relief valves once pressure was relieved. In addition, damage to the relief valve discharge piping to the HUT did not impact the ability of the relief valves or PORVs from performing their LTOP function. Therefore, the inspectors determined the finding was of very low safety significance (Green).

The inspectors concluded this finding did not have a cross-cutting aspect.

As the Document Signer, I reviewed Mr. O'Dwyer's non-concurrence as well as Ms. Stone's supervisory/Document Reviewer response. As part of my review, Jared Heck and I interviewed Mr. O'Dwyer to better understand his concerns, and I reviewed the draft Byron inspection report and provided background materials. I also interviewed Ms. Stone and Mr. Neurauter to fully understand his disposition of the issue involving water hammer. I also consulted Reactor Oversight Program policy documentation such as the applicable portions of Manual Chapter 0609, Significance Determination Process, and Manual Chapter 0612, Inspection Reports.

Based on my review, I agreed with the significance determination for the three performance deficiencies as described in the report (related to an RHR relief valve relieving to the HUT under normal plant conditions). The performance deficiencies involved the 2 green findings with associated Non-cited violations and the minor finding. I did request, however, that Ms. Stone provide an assessment of the two green findings against MC 0609, Appendix G, Shutdown Operations. She completed this assessment which also determined the performance deficiencies to be green, and she provided a revised inspection report write-up.

Although I agree with the significance determination for the findings, I do not agree with issuing the proposed Unresolved Item. The Unresolved Item involves a number of questions and issues raised by Mr. O'Dwyer associated with the adequacy of the licensee's establishment of a minimum HUT level as corrective actions for the NRC identified HUT design deficiency and other aspects of the HUT design (e.g., review of operation experience, alternate scenarios). These issues were raised in the proposed inspection report, but have not yet been raised to the licensee as part of an inspection. Manual Chapter 0612 presumes that prior to opening an Unresolved Item that an inspection has occurred, and prescribes that if an Unresolved Item is documented in an Inspection Report the Agency's and/or licensee's future action is clearly stated. As such, I find that documenting questions/issues for which there has been no substantiation or licensee engagement inconsistent with policy. The questions raised should be inspected prior to issuance of the inspection report.

In summary, I have concluded not to issue the inspection report, and to hold disposition of the described performance deficiencies pending additional inspection. The issues to be inspected include: the questions described in the inspection report input and associated questions regarding the adequacy of the licensee's corrective actions to the HUT design deficiencies as well as the following items not documented in the report: PORV set point adequacy, ASME code compliance, and questions associated with the alternate source term.

The non-concurrence will remain open pending completion and documentation of the inspection.

Anne T. Boland, Deputy Director
Division of Reactor Safety, Region III

UNITED STATES
NUCLEAR REGULATORY COMMISSION
LISLE, ILLINOIS 60532

NOT ISSUED

(See ML083180104 for final report IR2008009)

MEMORANDUM TO: B. Bartlett
Senior Resident Inspector
Byron Station

FROM: A. M. Stone, Chief
Engineering Branch 2
Division of Reactor Safety

SUBJECT: BYRON STATION
DRS INPUT TO INTEGRATED REPORT 05000454/2008003; 05000455/2008003

Enclosed is the report input for the Byron Station, Inspection Report 05000454/2008003; 05000455/2008003. This report input documents the results of an inspection of several corrective action documents addressing the hold-up tanks. I have reviewed this input and ensured compliance with Inspection Manual Chapter (IMC) 0612 including confirming each finding was reviewed for potential cross-cutting aspects. This input is ready for inclusion into the integrated report and dissemination to the public.

Enclosure: Input to Inspection Report 05000454/2008003; 05000455/2008003

cc w/encl: R. Skokowski, Branch Chief
S. Eldridge, Site Secretary

CONTACT: Ann Marie Stone, DRS
(630) 829-9729

DOCUMENT NAME: G:\DRS\Workinprogress\BYRON Input to DRP Report 2008-003 HUT.doc
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OFFICE	RIII		RIII	N		
NAME	GO'Dwyer: Is Non-concur (see Form 757)		AMStone			
DATE	11/ /08		11/ 08			

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Cover Letter

_X_ Two very low significant findings with associated Non-Cited Violations were identified.

TITLE PAGE

Inspectors: G. O'Dwyer, Reactor Engineer

SUMMARY OF FINDINGS

A. NRC-Identified and Self-Revealed Findings

Cornerstone: Barrier Integrity

- Green. A finding of very low safety significance and associated Non-Cited Violation (NCV) of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," having very low safety significance (Green) was identified by the inspectors in that the licensee failed to correctly analyze the consequences of discharges to the Boron Recycle Holdup Tank (HUT) of the residual heat removal (RH) suction and discharge relief valves. Specifically, the licensee failed to recognize that RH relief valve discharges could be as hot as 350 degrees F and be of sufficient volume to fail the HUTs catastrophically by over-pressurization; and that minimum HUT water level controls were required to ensure adequate quenching to prevent the HUTs from failing. The licensee took immediate corrective actions, which included requiring a minimum 40 percent HUT level whenever the RH relief valves were lined up to the HUT.

  The finding was more than minor because, if left uncorrected, the issue would have become a more significant safety concern. The inspectors determined that the lack of HUT minimum level controls to ensure adequate discharge quenching only degraded the ability of the HUTs to maintain the Radiological Barrier Functionality of the Auxiliary Building of a PWR, which was a finding having very low safety significance (Green). The finding was associated with the design control and procedure quality attributes of the cornerstone. The inspectors determined that the finding did not have a cross-cutting aspect. (Section 4OA2.2.b.1)

- Green. A finding of very low safety significance and associated NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," having very low safety significance (Green) was identified by the inspectors in that the licensee failed to correctly analyze the potential for water hammer from discharges of relief valves that discharge to the HUT. The licensee took immediate corrective actions of preliminarily analyzing the potential for water hammer, establishing a minimum HUT level, and entering the issue into the corrective action program.

  The finding was more than minor because, if left uncorrected, the issue would have become a more significant safety concern. The inspectors determined that the lack of analyses for potential water hammer from the RH relief valves into the HUT only degraded the assurance of the ability of the HUTs to maintain the Radiological Barrier Functionality of the Auxiliary Building of a PWR, which was a finding having very low safety significance (Green). The finding was associated with the design control attribute of the cornerstone. The inspectors determined that the finding did not have a cross-cutting aspect. (Section 4OA2.2.b.2)

B. Licensee-Identified Violations

No violations of significance were identified.

REPORT DETAIL

4. OTHER ACTIVITIES

4OA2 Identification and Resolution of Problems (71152)

.2 Annual Sample Review

a. Inspection Scope

The inspectors completed one inspection sample regarding problem identification and resolution by conducting in-depth reviews for the following condition reports:

- Byron AR 622574, "Concerns regarding relief valves and inputs into the HUT," April 27, 2007;
- Byron AR 624329, "Hot Leg Check Valve Leak-By Indication," initiated May 2, 2007;
- Byron AR 649710, "Potential Vulnerability with RH suction relief valve to HUT," July 13, 2007;
- Byron AR 679954, "Recycle Holdup Tank Level Administrative Controls," October 4, 2007; and
- Byron AR 680626, "NRC Potential Green Finding and Associated NCV – HUT Level," October 4, 2007.

The inspectors reviewed the licensee's actions to verify whether: (1) the problems were accurately identified; (2) the causes were adequately ascertained; (3) extent of condition and generic implications were appropriately addressed; (4) previous occurrences were considered; and (5) corrective actions proposed/implemented were appropriately focused to address the problems and were commensurate with the safety significance of the issues. The inspectors, however, could not determine the effectiveness of the corrective actions because most of the corrective actions were still in planning stage. The inspectors also reviewed other documents including applicable sections of the Byron/Braidwood Updated Final Safety Analysis Report (UFSAR), Chapters 3, 5, 6, 9, 11, and 15; Technical Specification (TS) 3.4.12 and its bases; plant startup and shutdown procedures; Augmented NRC Inspection Report 05000456/90-20; Calculations CN-CRA-00-47, "Braidwood/Byron Doses from Recycle Holdup Tanks," and "Spent Resin Tank Failures" and CN-CRA-07-50; September 15, 2007; "Byron/Braidwood HUT Response to Opening of the RHR Relief Valve"; NUREG-1326, Regulatory Analysis for the Resolution of Generic Issue 94, "Additional Low-Temperature Overpressure Protection for Light-Water Reactors," December 1989; Information Notice (IN) 90-05, "Intersystem Discharge of Reactor Coolant"; Information Notice (IN) 92-36, "Intersystem LOCA Outside Containment"; and plant data records.

b. Findings

(1) Failure to Have Proper HUT Level Controls

Introduction: The inspectors identified an NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," having very low safety significance (Green), in that the licensee failed to correctly analyze the consequences of discharges to the Boron Recycle Holdup Tank (HUT) of the residual heat removal (RH) suction and discharge relief valves. Specifically, the licensee failed to recognize that RH discharge and suction relief valve discharges could be as high as 350ºF and be of sufficient volume to fail both HUTs catastrophically by over-pressurization; and that minimum HUT water level controls were required to ensure adequate quenching to prevent the HUTs from failing catastrophically.

Description: The inspectors noted that the RH suction and discharge relief valves were originally designed to discharge to a sparger pipe with special nozzles in the PRT that was designed to discharge below the PRT water level in order to ensure adequate quenching. The PRT is designed for 100 psig and 200ºF with a cooling system if PRT temperatures exceed 120ºF during or after a discharge. The UFSAR Section 5.4.11.3 stated that "If a discharge results in a pressure that exceeds the design, the rupture discs on the tank would pass the discharge through to the containment." During construction the Architect-Engineer changed the discharge piping design and rerouted the piping so that the RH suction relief valves go to the HUTs instead of the PRT. The discharge piping enters the HUT with no sparger. The inlet piping entered the HUT at about 35 percent level; however, the minimum level in which the licensee operated could be as low as 5 percent, which did not provide any quenching. The licensee procedures usually maintained the HUT water level below the inlet piping entrance when the RH relief valves were lined up to discharge to the HUT. There were many times during plant life where the RH relief valves were lined up to discharge to the HUT while the plant was in Mode 3 and 4, i.e., the discharges would have been over 212°F. This configuration prevented the water from quenching the steam and hot water discharge from the reliefs. The inspectors were concerned that routing to the HUT could result in failure of the HUT during a lift of the relief valves when HUT level was low. Specifically, the HUT was designed for only 15 psig and 200°F as documented in UFSAR, Revision 11, Section 15.7.2.1. Inadequately quenched steam would cause a drastic increase in pressure resulting in tank failure.

The licensee performed a calculation, CN-CRA-07-50, to determine the response of one RH suction relief valve discharging to one HUT for two different cases. The first case assumed an initial level of 5-feet below the inlet nozzle of the RH suction relief valve discharge piping to the HUT and the second assumed 5-feet above the nozzle. The calculation concluded that the HUT would have failed catastrophically within about 50 seconds if the HUT level had been below the inlet nozzle which would have prevented the discharge from being quenched at all. The HUT failure would release the contents of the HUT to the HUT room in the Auxiliary Building. The calculation concluded that if the initial HUT level was 5-feet over the inlet nozzle the HUT air space pressure would not exceed the HUT design pressure of 15 psig even if the RH suction relief valve was open for 30 minutes. The calculation did not attempt to determine the minimum level corresponding to an acceptable HUT response. Licensee's calculation, CN-CRA-08-9, was completed February 14, 2008, and provided quantitative analysis supporting the licensee's previous engineering judgment that 40 percent HUT level provided sufficient quenching of hot discharges to prevent catastrophic HUT failure. The inspectors had concerns with deficiencies in calculations CN-CRA-07-50 and CN-CRA-08-9 that reduced the conservatism and the concerns are stated in and will be followed up by the Unresolved Item in Section (b)3 below.

Title 10 CFR Part 50, Appendix A, GDC 34 required the RHR system to have enough redundancy to perform its safety function with a loss of offsite power and assuming a single failure. The inspectors noted that UFSAR Section 5.4.7.2.6.c "Residual Heat Removal System" stated "Byron/Braidwood are licensed as "hot shutdown" or Class 2 Plant in accordance with Branch Technical Position RSB 5-1." Branch Technical Position (BTP) RSB 5-1 (dated July 1981), stated in Section A.2.4 that the RHR system shall bring the reactor to cold shutdown without offsite power assuming the most limiting single failure. The inspectors noted that the licensee committed to meet the requirements of BTP 5-1, July 1981, in the application for a license. Section C.2 of BTP 5-1 required that "Fluid discharged through the RHR system pressure relief valves must be collected and contained such that a stuck open relief valve will not result in a non-isolatable situation in which the water provided to the RCS to maintain the core in a safe condition is discharged outside of containment." Technical Specification Bases Section 3.4.12 stated "The RHR suction relief valves are considered active components. Thus, the failure of one valve is assumed to represent the worst case single active failure." The inspectors concluded that the licensee needed to consider the potential for an RH relief valve to lift as designed.

The inspectors noted that UFSAR Section 3.9.1.1.b defined Upset Conditions (Incidents of Moderate Frequency) as "any deviations from normal conditions anticipated to occur often enough that design should include a capability to withstand the conditions without operational impairment. The upset conditions include those transients which result from any single operator error or control malfunction. Upset conditions include any abnormal incidents not resulting in a forced outage and also forced outages for which the corrective action does not include any repair of mechanical damage." Cold over-pressurization events are listed as Upset Conditions. Section 3.9.1.1.b further stated that "RCS cold over-pressurization occurs during startup and shutdown conditions at low temperature, with or without existence of a steam bubble in the pressurizer, and is especially severe when the reactor coolant system is in a water-solid configuration. The event is inadvertent, and usually generated by any one of a variety of malfunctions or operator errors." Based on the discussion in the UFSAR, the inspectors concluded that the lifting of a relief valve was an expected occurrence and that this event should not result in damage to the HUT.

The two applicable accident analyses relate to the atmospheric and ground release of the HUT. The UFSAR Section 15.7.2 analyzed the worst case radioactive atmospheric release and assumed that the postulated event was initiated by cracks in the HUT and operator error. The analysis assumed that this accident is expected to occur with the frequency of a limiting fault. The UFSAR Section 15.0.1.4 defined Limiting Fault as an event not expected to occur; however, UFSAR Section 3.9.1.1.b indicated that lifting of an RH relief valve was an expected occurrence and that this event should not result in damage to the HUT and certainly not catastrophic failure. The analysis also assumed that only one HUT would fail, however before the finding both RH reliefs had been lined up to both HUTs at various times and therefore discharges in excess of 265ºF could have been relieved to both HUTs causing both HUTS to rupture. The inspectors concluded that failure to assume two HUTs rupturing had an impact on the calculated release rates; however, the margin contained in the calculation because of other conservatisms was sufficient. The inspectors noted in UFSAR Section 15.7.3.1 that the postulated radioactive ground release was analyzed assuming an unexpected and uncontrolled rupture of the HUT. The possibility of failure of the HUT was considered a limiting fault and not expected to occur. Since low temperature/over pressure (LTOP) events could cause HUT failures, the inspectors concluded that an initial low level in the HUTs would increase the probability of rupturing the HUTs beyond that assumed in the UFSAR.

With respect to Mode 4 conditions (reactor coolant from 350ºF to 212ºF, the licensee stated that the administrative controls in place during Mode 4 would prevent lifting of the relief valves. The inspectors noted that these actions were initiated in response to several Information Notices as well as Generic Letter 90-06, "Resolution of Generic Issues 70, 'PORV and Block Valve Reliability' and 94, 'Additional LTOP Protection for PWRs.'" While the inspectors recognized that administrative controls would greatly reduce the probability of a relief valve lifting, the inspectors concluded that the licensee needed to consider the potential for a relief valve to lift as designed. Specifically, TS 3.4.12.c and d requires the RH relief valve to be OPERABLE in Mode 4, regardless of the presence of a pressurizer bubble, when the relief valves are being used for LTOP purposes. The relief valves are required to lift prior to 450 psig and, as discussed in UFSAR Section 5.2.2.11.2, an RH suction relief valve is sized to relieve the capacity of one charging pump. The inspectors also noted that even if the licensee was using the pressurizer power operated relief valves (PORVs) as the TS-required LTOP pathway, the RH relief valves would lift prior to the PORVs reaching their setpoint because the RH relief valves lift at a lower pressure and would be in service during most of Mode 4 and all of Mode 5. Plant procedures did not instruct the licensee to gag the RH relief valves when the valves were not used for LTOP.

The inspectors noted that the licensee had experienced actual lifting of the RH suction relief valves during heatup conditions: Braidwood on December 1, 1989, and Byron on April 21, 2001, and May 1, 2007. While the Braidwood event occurred with the RCS water-solid and resulted in an increase in administrative controls industry-wide, the events at Byron demonstrated the possibility of RH relief valve lifts when operating near the transition of Mode 4 and 3 with the RH system not in service. Finally, the inspectors noted that in Calculation CN-CRA-07-50, the licensee determined the response of the HUT assuming initial levels of 5-feet above and 5-feet below the discharge nozzle of the RH relief valves. The licensee did not determine the minimum level corresponding to an acceptable HUT response.

Analysis: The inspectors determined that the failure to fully analyze consequences of the lifting of RH suction relief valves which were expected occurrences and the failure to ensure adequate minimum levels in the HUT to prevent resulting damage to the HUT were a performance deficiency warranting a significance evaluation. The inspectors determined the finding was more than minor in accordance with IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Disposition Screening," because the finding was associated with the Barrier Integrity cornerstone and maintaining the Radiological Barrier Functionality of the Auxiliary Building of a PWR. The finding was associated with the design control and procedure quality attributes of the cornerstone. Specifically, the licensee did not appropriately ensure that design analyses bounded potential RH relief valve discharges greater than 212ºF to the HUTs. Also, design requirements were not translated into licensee procedures to appropriately ensure a minimum HUT water level to ensure adequate quenching to prevent potential catastrophic HUT failures due to over-pressurization.

The inspectors evaluated the finding using IMC 0609, Attachment 0609.04, "Phase 1 – Initial Screening and Characterizations of Findings." The inspectors determined in

Tables 2 and 4a of the attachment that the lack of procedural minimum levels for adequate quenching in the HUTs only degraded the ability of the HUTs to maintain the Radiological Barrier Functionality of the Auxiliary Building of a PWR; therefore, the finding screened as having very low safety significance (Green). The inspectors concluded this finding did not have a cross-cutting aspect.

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion III, "Design Control," requires, in part, that design control measures shall provide for verifying the adequacy of design and that the design basis is correctly translated into procedures and instructions.

Contrary to the above, from plant construction to October 4, 2007, the licensee failed to incorporate appropriate minimum HUT level requirements into the HUT level control procedures. However, because this issue was of very low safety significance, and it was entered into the licensee's corrective action program (Byron ARs 679954 and 680626), this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy (NCV 05000454/2008003-XX, NCV 05000455/2008003-XX (DRS)).

(2) Failure to Analyze Potential Water Hammer

Introduction: The inspectors identified an NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," having very low safety significance (Green), in that the licensee failed to correctly analyze the potential for water hammer from discharges of relief valves that discharge to the HUT.

Description: The inspectors were concerned that after a discharge from the RH relief valves, steam would exist in the approximately 100-feet of 4-inch piping between the RH suction relief valves and the HUT. The inspectors were concerned that if the water level in the HUT was above the connection from the relief line, as steam condenses in the piping, water would be drawn up into the piping. Subsequent relief valve lifts of any of the approximately 20 other relief valves that discharge through the same piping to the HUTs could cause a water hammer and piping failure.

In response, the licensee determined that the existing piping stress analysis from the RH suction relief valves to the HUT considered the reaction loads associated with actuation of the relief valves. In addition, the licensee stated that the potential for condensation-induced water hammer was low due to: (1) the approximate 30-feet elevation change in the piping between the tank and the RH relief valves which would limit the driving pressure available to move a slug backwards in the line; and (2) an extended blowdown into the tank would heat the water in the vicinity of the nozzle thus reducing the condensation rate and limiting the reduction in void pressure, thereby limiting the available driving pressure. The licensee noted that the pipe routing from the suction relief valves to the HUTs did contain loop seals that would allow for fluid to collect in low points forming columns available for potential water hammer.

Although the discharge piping was not evaluated for the transient forces due to the dynamic effect of acceleration of these water columns, the licensee anticipated that these loads would be relatively low. The licensee indicated that the relief valve discharge piping was seismically supported, and since a seismic event was not postulated to occur simultaneously with a relief valve discharge, the predicted seismic loads on the supports provided margin to accommodate the transient loads. Furthermore, the licensee indicated that Braidwood Unit 1 had experienced a lift of the

RH suction relief valve and noted that no damage occurred to pipe and pipe supports. Based on this operating experience, the licensee postulated that resulting transient loads on piping and pipe supports would have a similar magnitude, and, therefore, concluded that any damage to piping and pipe supports would be minimal.

The inspectors reviewed the licensee's response and isometric drawings for the system, and conducted interviews with licensee personnel. The inspectors determined that after a discharge from the RH relief valves, steam might exist in the 4-inch piping between the RH suction relief and the HUT. The inspectors determined that, if the water level in the HUT was above the connection from the relief line, as steam condensed in the piping, water would be drawn up in into the piping, potentially causing a water hammer when other relief valves lift.

The inspectors also noted that although existing seismic supports would provide margin to accommodate transient loads in the direction of the support restraint, the postulated dynamic transient might create piping forces in directions that were not restrained by the existing pipe supports. The licensee entered the discharge line water hammer concern into the corrective action program as Issue Report 680626 (Byron). The licensee also implemented compensatory actions, including raising the water level in the HUT to above the discharge line connection at the HUT nozzle in order to ensure that quenching would occur.

The inspectors determined that the above interim compensatory action would condense steam discharged into the HUT and reduce the likelihood of HUT over-pressurization, minimize the driving pressure available to move a water slug backwards into the discharge line as steam in the line condensed, and lower the potential for a condensation-induced water hammer.

Analysis: The inspectors determined that the failure to evaluate the potential for water hammers was a performance deficiency warranting a significance evaluation. The inspectors determined the finding was more than minor in accordance with IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Disposition Screening," because the finding was associated with the Barrier Integrity cornerstone and maintaining the Radiological Barrier Functionality of the Auxiliary Building of a PWR. The finding was associated with the design control and procedure quality attributes of the cornerstone. Specifically, the licensee did not appropriately evaluate potential water hammer from the relief valves discharging to the HUT per design.

The inspectors evaluated the finding using IMC 0609, Attachment 0609.04, "Phase 1 - Initial Screening and Characterizations of Findings." The inspectors determined in Tables 2 and 4a of the attachment that the lack of analyses for potential water hammer from the relief valves into the HUT only degraded the assurance of the ability of the HUTs to maintain the Radiological Barrier Functionality of the Auxiliary Building of a PWR and Table 4a screened the finding as having very low safety significance (Green). The inspectors concluded this finding did not have a cross-cutting aspect.

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion III, "Design Control," requires, in part, that design control measures shall provide for verifying the adequacy of design and that the design basis is correctly translated into procedures and instructions. Contrary to the above, from plant construction to October 4, 2007, the licensee failed to appropriately evaluate potential water hammer from the relief valves discharging to the

HUT per design. However, because this issue was of very low safety significance, and it was entered into the licensee's corrective action program (Byron AR 679954 and AR 680626), this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy (NCV 05000454/2008003-XX, NCV 05000455/2008003-XX (DRS)).

(3) Potential Inadequate Corrective Actions to Address HUT Issue

Introduction: The inspectors identified an unresolved item regarding potential inadequate corrective actions taken to address the above violations concerning the RH relief valves discharging to the HUT.

Description: As a result of the above NCVs, on October 4, 2007, the licensee initiated compensatory measures that required at least a minimum HUT level of 40 percent be maintained when the RH discharge and suction relief valves are aligned to discharge to the HUT in Mode 4. The licensee also required that the RH discharge and suction relief valves be aligned to only one HUT at a time. The licensee chose the 40 percent based on engineering judgment with no supporting quantitative analysis. As stated above, in calculation CN-CRA-07-50, the licensee determined the response of the HUT to a RH suction relief valve discharge for two different cases, the first assuming initial levels of 5-feet above and the second assuming 5-feet below the discharge nozzle of the RH relief valves. The licensee did not determine quantitatively the minimum level corresponding to an acceptable HUT response. The inspectors requested supporting analysis for the licensee's engineering judgment. The licensee completed calculation CN-CRA-08-9, Revision 0, "Byron/Braidwood RHUT Response to Opening of the RHR Relief Valve," on February 14, 2008. The licensee provided calculation CN-CRA-08-9 to the NRC as the basis for the minimum HUT water level of 40 percent. The calculation concluded that the 40 percent level would prevent HUT failure during any design conditions. Calculation CN-CRA-08-9 was based on the analyses of calculation CN-CRA-07-50. The inspectors had the following concerns:

- Calculations CN-CRA-07-50 and CN-CRA-08-9 did not adequately detail the methodology; the different event analyses were not adequately defined, and the same event analyses appear to have been designated multiple names without sufficient correlation between the different names. Some examples of the lack of sufficient documentation were the calculations discussed different cases, test runs, computer runs, test cases, etc. with inadequate definitions of each designation and inadequate correlation between the designations.

- The calculation CN-CRA-07-50 assumed perfect mixing of the entire water volume of the HUT with the HUT water locally around the steam discharge as the steam discharge rises through the water layer above the inlet piping. The inspectors noted that "perfect" mixing is impossible; therefore, imperfect mixing was possible. Additional information is needed to determine the impact of assuming non-perfect mixing.

- Calculations CN-CRA-07-50 and CN-CRA-08-9 assumed that the maximum RH suction relief valve flowrate during design-required conditions was 475 gpm. The Technical Specifications and supporting analyses required that the RH suction relief flowrate be a minimum of 475 gpm at 350°F. The inspectors questioned whether the licensee used a bounding value of discharge flowrate of the RH suction valve in the calculation and analyses. Additionally, the inspectors also noted that two RH systems are typically used during Shutdown Cooling in Mode 4 and therefore the flowrate should consider the potential for 2 RH suction relief and 2 RH discharge relief valves being opened.

- Calculation CN-CRA-07-50 assumed perfect mixing of the steam with the air-gas mixture in the HUT gas space. The inspectors noted that "perfect" mixing is impossible; therefore, imperfect mixing was possible. Additional information is needed to determine the impact of assuming non-perfect mixing.

- Calculation CN-CRA-08-9 used the design pressure for the bottom of the HUT (15 psig + 31 feet 6 inches of water hydrostatic head) as the failure pressure for the HUT gas-space instead of the HUT gas-space design pressure of 15 psig. The inspector was concerned that this would overestimate the pressure at which the upper portion of the HUT would fail and significantly underestimate the minimum HUT level required to ensure quenching.

- Section 4.6, "Inputs" section of Calculation CN-CRA-08-9 stated that Run 1 of the calculation assumed that the initial HUT air-gas space is filled with 100 percent vapor. The inspectors noted that the HUT atmosphere would likely be an air-gas mixture. Assuming 100 percent water vapor initially in the air-gas space would significantly underestimate the final HUT pressure because the initial pressure of 100 percent water vapor at 115ºF would be approximately 1.5 psia. Assuming 100 percent water vapor would also underestimate the final HUT pressure because the HUT initial air-gas mixture is composed of non-condensable gases, which would contribute a higher pressure when heated than 100 percent water vapor.

- The licensee had not completed their corrective actions with respect to analysis of the impact of other valves releasing to the HUT including the CVCS and SI system suction relief valves when in the piggyback mode on the RH system after a LOCA. Therefore, the inspectors were unable to assess resulting analysis or corrective actions.

- The Architect-Engineer rerouted the RH suction and discharge relief valves' piping from the PRT to the HUT and did not require a minimum HUT level to ensure quenching to be maintained. It is not clear whether the licensee evaluated this as a defect as defined in 10 CFR Part 21 and should be reported as required by 10 CFR Part 21.

- From the limited review of the licensee's corrective actions, it is not clear if the licensee's extent of condition considered whether discharge of other valves had been improperly rerouted to the HUT instead of to the Pressurizer Relief Tank (PRT), Reactor Coolant Drain Tank (RCDT), or Volume Control Tank (VCT).

- In Byron's IR 622574, the licensee proposed to go solid in Mode 4 during Braidwood's refueling outage A1R13. The inspectors were concerned that operating solid in Mode 4 (RCS temperature from 350 – 200ºF would drastically increase the probability of an LTOP overpressure event and increase the probability of lifting the RH suction and discharge relief valves.

- On April 21, 2001, and May 1, 2007, the RH suction relief valve discharged to the Byron HUTs unexpectedly during ECCS check valves testing due to procedure deficiencies (IR 624329). The inspectors were concerned that the corrective actions for these events had not been entered into the licensee's Corrective Action Program.

- It is unclear whether the inputs used in Section 6.2 of calculation CN-CRA-07-50 were verified or received a peer check. In addition, it is unclear whether calculation CN-CRA-08-9 has been approved as a design base calculation.

This issue is considered unresolved (URI 05000454/2008003-XX, URI 05000455/ 2008003-XX (DRS)) pending review of the calculations, corrective actions and other information needed from the licensee.

(4) Analysis Did Not Bound Operating Conditions

The inspectors noted that the existing UFSAR analysis for the rupture of a recycle holdup tank failed to recognize that the gas spaces of the HUTs were normally cross-connected and that a Gas Decay Tank normally had open communication with at least one HUT. The licensee stated that based on conservative assumptions in the analysis, the actual plant configuration (Gas Decay tank providing cover gas to two HUTS) was bounded. Specifically, the calculated dose for a postulated recycle HUT failure was based on the following:

- The assumed inventory of noble gases in the HUT was based on transferring the total inventory of primary coolant from one unit at maximum purification letdown flow;

- No removal of noble gas was assumed in the purification letdown flow; and

- When the tank failure occurred, a portion of the iodine in the water and all of the noble gas activity was assumed to become airborne and released to the environment.

The inspectors reviewed Calculation CN-CRA-00-47, "Braidwood/Byron Doses from Recycle Holdup Tanks and Spent Resin Tank Failures," dated June 7, 2000, and noted that Calculation CN-CRA-00-47 assumed the HUT was initially filled to 80 percent capacity, the water contents of the tank would be released in 5-minutes, and the HUT was isolated from the other HUT and Gas Decay tank. These assumptions were consistent with those in UFSAR Section 15.7.2, which stated that the postulated events that could cause the worst case radionuclide inventory were cracks in the HUT and operator error. The inspectors concluded that the calculation of record, CN-CRA-00-47, did not account for the actual plant conditions or the failure mechanism described in Section 4OA2.2.b.1 of this report. Specifically the initial level of 80 percent was not consistent with current operation because HUT level could be as low as 5 percent. A crack in the HUT and subsequent 5-minute release of water content was not consistent with the expected rupture of the HUT with HUT water level below the relief valve discharge line.

Although the calculation contained conservative assumptions with respect to gaseous releases, it did not specifically address how these assumptions bound the actual plant configuration and/or operation of the waste gas systems. The inspectors reviewed the calculation and determined these discrepancies (i.e., quicker release rate, lower initial volume, and cross-connected gas systems) would have an impact on the calculated release rates; however, the margin contained in the calculation was sufficient.

The inspectors concluded that the failure to account for actual plant configurations in an accident analysis was a violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control." The inspectors assessed this violation in accordance with IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening," and determined that the finding was minor because the available margin and other conservative assumptions were sufficient to compensate for identified discrepancies. Therefore, in accordance with IMC 0612, this violation of minor significance is not subject to enforcement action in accordance with Section IV of the NRC's Enforcement Policy.

SUPPLEMENTAL INFORMATION

KEY POINT OF CONTACT

Licensee

None

Nuclear Regulatory Commission

None

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000454/2008003-xx, 05000455/2008003-xx (DRS)	NCV	Failure to Have Proper HUT Level Controls
05000454/2008003-xx, 05000455/2008003-xx (DRS)	NCV	Failure to Analyze Potential Water Hammer
05000454/2008003-xx, 05000455/2008003-xx (DRS)	URI	Potential Inadequate Corrective Actions to Address HUT Issue

Closed

05000454/2008003-xx, 05000455/2008003-xx (DRS)	NCV	Failure to Have Proper HUT Level Controls
05000454/2008003-xx, 05000455/2008003-xx (DRS)	NCV	Failure to Analyze Potential Water Hammer

LIST OF DOCUMENTS REVIEWED

The following is a list of documents reviewed during the inspection. Inclusion on this list does not imply that the NRC inspectors reviewed the documents in their entirety, but rather, that selected sections of portions of the documents were evaluated as part of the overall inspection effort. Inclusion of a document on this list does not imply NRC acceptance of the document or any part of it, unless this is stated in the body of the inspection report.

4OA2 Identification and Resolution of Problems (71152)

Byron AR 622574; "Concerns regarding relief valves and inputs into the HUT"; initiated April 27, 2007

Byron AR 624329; "Hot Leg Check Valve Leak-By Indication"; initiated May 2, 2007

Byron AR 649710; "Potential Vulnerability with RH suction relief valve to HUT"; initiated July 13, 2007

Braidwood AR 649581; "Potential Vulnerability with RH Suction Relief Disch to Hut"; initiated July 12, 2007

Braidwood AR 677075; "Recycle Hold Up Tank Level Administrative Controls"; initiated September 28, 2007

Byron AR 679954; "Recycle Hold Up Tank Level Administrative Controls"; initiated October 4, 2007

Byron AR 680626; "NRC Potential Green Finding and Associated NCV – HUT level"; initiated October 5, 2007

Braidwood Procedure 1BwGP 100-5; "Plant Shutdown and Cooldown"; Revision 34

Byron Procedure 1BGP 100-5; "Plant Shutdown and Cooldown"; Revision 52

Calculation CN-CRA-00-47; "Braidwood/Byron Doses from Recycle Holdup Tanks and Spent Resin Tank Failures"; June 7, 2000

CN-CRA-07-50; "Byron/Braidwood HUT Response to Opening of the RHR Relief Valve"; September 25, 2007

Byron/Braidwood Unit 1 Drawing M-550; "Aux Bldg. El. 346' and 364' Boric Acid Processing"; Sheet 3; Revision P; and Sheet 4; Revision K

Chicago Bridge and Iron Drawing 62573; "27' x 24' 9" High Ellip Roof Tk"; dated March 1, 1977

Appropriate Piping Isometric Southwest Fabricating and Welding Drawings for the Boric Acid Processing System of Byron Station Unit 1 and 2

INSPECTION PLAN

FACILITY: BYRON STATION UNIT 1 AND 2

INSPECTION REPORT NUMBER: 50-454/08-009; 50-455/08-009;

ONSITE INSPECTION DATES: October 6 -10 ,2008 (Entrance Meeting 6th and Exit Meeting 10th)

RESIDENT NOTIFIED: Ray Ng, Acting SRI

INSPECTORS: G. O'Dwyer, M. Holmberg

INSPECTION PROCEDURE(S): IP-71152, Identification and Resolution of Problems; Section 03.02 (one sample)

INSPECTION TYPE: Baseline

OBJECTIVE(S): Follow-up of previously identified compliance issues.

Prepared By: M Holmberg 9/22/2008
Inspector -M. Holmberg

Concurred By Gerard O'Dwyer 9/22/08
Inspector G. O'Dwyer

Approved By: M Holmberg for David Hills per E-mail 9/24/08
DRS Branch Chief - D. Hills

Reviewed By: R A Skokowski
DRP Branch Chief - R. Skokowski

I) BACKGROUND

NRC compliance issues were identified during review of concerns identified in Byron Allegation RIII-2007-A-024. During this review, three NRC findings were identified as documented in this Allegation file. These findings were communicated with the licensee on October 4, 2007. The concerns for this Allegation were reviewed and closed. The closure letter to the CI was issued on October 19, 2007.

During review of documents above (the licensee's PTLR report), a potential discrepancy for plant operation above 350 degrees F associated with the PORVs for LTOP was identified. This issue was resolved during inspection preparation to the satisfaction of the inspectors, and no further reviews are planned for this area.

II) INSPECTION ACTIVITIES (IP 71152) (Charge Codes BIP, BI and BID)

A) Inspection Objectives

1) Document the findings associated with the Boron Recycle Holdup Tank (HUT) identified in Allegation file RIII- 2007-A-024.

2) Review licensee corrective actions for these findings and document the results of this review.

B) Inspection Scope

The inspectors will perform a review in accordance with Sections 02.02 and 03.02 of IP 71152 which require 4-7 issues per year be selected for in-depth review, as necessary to verify that the licensee has taken corrective actions commensurate with the safety significance.

B.1) The inspectors will document and perform an in-depth review for the licensee completed or planned corrective actions associated with the following findings identified in Byron allegation file RIII- 2007-A-024 **(Mel has lead):**

i) A finding of very low safety significance and associated Non-Cited Violation (NCV) of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," having very low safety significance (Green) was identified by the inspectors in that the licensee failed to correctly analyze the consequences of discharges to the Boron Recycle Holdup Tank (HUT) of the residual heat removal (RH) suction and discharge relief valves. Specifically, the licensee failed to recognize that RH relief valve discharges could be as hot as 350 degrees F and be of sufficient volume to fail the HUTs catastrophically by over-pressurization; and that minimum HUT water level controls were required to ensure adequate quenching to prevent the HUTs from failing. The licensee took immediate corrective actions, which included requiring a minimum 40 percent HUT level whenever the RH relief valves were lined up to the HUT.

II. <u>INSPECTION ACTIVITIES (IP 71152) -Continued</u>

B.1) Inspection Scope - Continued

ii) A finding of very low safety significance and associated NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," having very low safety significance (Green) was identified by the inspectors in that the licensee failed to correctly analyze the potential for water hammer from discharges of relief valves that discharge to the HUT. The licensee took immediate corrective actions of preliminarily analyzing the potential for water hammer, establishing a minimum HUT level, and entering the issue into the corrective action program.

B.2) The inspectors will confirm that the licensee has captured and initiated appropriate corrective actions for the following concerns identified during review of Calculations CN-CRA-08-9 and CN-CRA-07-50 associated with the HUT tank response to opening of the RHR suction pipe relief valve **(Gerry has lead):**

- Calculations CN-CRA-07-50 and CN-CRA-08-9 did not adequately detail the methodology; the different event analyses were not adequately defined, and the same event analyses appear to have been designated multiple names without sufficient correlation between the different names. Some examples of the lack of sufficient documentation were the calculations discussed different cases, test runs, computer runs, test cases, etc. with inadequate definitions of each designation and inadequate correlation between the designations.

- The calculation CN-CRA-07-50 assumed perfect mixing of the entire water volume of the HUT with the HUT water locally around the steam discharge as the steam discharge rises through the water layer above the inlet piping. The inspectors noted that "perfect" mixing is impossible; therefore, imperfect mixing was possible. Additional information is needed to determine the impact of assuming non-perfect mixing.

- Calculations CN-CRA-07-50 and CN-CRA-08-9 assumed that the maximum RH suction relief valve flowrate during design-required conditions was 475 gpm. The Technical Specifications and supporting analyses required that the RH suction relief flowrate be a minimum of 475 gpm at 350ºF. The inspectors questioned whether the licensee used a bounding value of discharge flowrate of the RH suction valve in the calculation and analyses. Additionally, the inspectors also noted that two RH systems are typically used during Shutdown Cooling in Mode 4 and therefore the flowrate should consider the potential for 2 RH suction relief and 2 RH discharge relief valves being opened.

- Calculation CN-CRA-07-50 assumed perfect mixing of the steam with the air-gas mixture in the HUT gas space. The inspectors noted that "perfect" mixing is impossible; therefore, imperfect mixing was possible. Additional information is needed to determine the impact of assuming non-perfect mixing.

II. INSPECTION ACTIVITIES (IP 71152) -Continued

B.2) Inspection Scope - Continued

- Calculation CN-CRA-08-9 used the design pressure for the bottom of the HUT (15 psig + 31 feet 6 inches of water hydrostatic head) as the failure pressure for the HUT gas-space instead of the HUT gas-space design pressure of 15 psig. The inspector was concerned that this would overestimate the pressure at which the upper portion of the HUT would fail and significantly underestimate the minimum HUT level required to ensure quenching.

- Section 4.6, "Inputs" section of Calculation CN-CRA-08-9 stated that Run 1 of the calculation assumed that the initial HUT air-gas space is filled with 100 percent vapor. The inspectors noted that the HUT atmosphere would likely be an air-gas mixture. Assuming 100 percent water vapor initially in the air-gas space would significantly underestimate the final HUT pressure because the initial pressure of 100 percent water vapor at 115ºF would be approximately 1.5 psia. Assuming 100 percent water vapor would also underestimate the final HUT pressure because the HUT initial air-gas mixture is composed of non-condensable gases, which would contribute a higher pressure when heated than 100 percent water vapor.

- The licensee had not completed their corrective actions with respect to analysis of the impact of other valves releasing to the HUT including the CVCS and SI system suction relief valves when in the piggyback mode on the RH system after a LOCA. Therefore, the inspectors were unable to assess resulting analysis or corrective actions.

- The Architect-Engineer rerouted the RH suction and discharge relief valves' piping from the PRT to the HUT and did not require a minimum HUT level to ensure quenching to be maintained. It is not clear whether the licensee evaluated this as a defect as defined in 10 CFR Part 21 and should be reported as required by 10 CFR Part 21.

- From the limited review of the licensee's corrective actions, it is not clear if the licensee's extent of condition considered whether discharge of other valves had been improperly rerouted to the HUT instead of to the Pressurizer Relief Tank (PRT), Reactor Coolant Drain Tank (RCDT), or Volume Control Tank (VCT).

- In Byron's IR 622574, the licensee proposed to go solid in Mode 4 during Braidwood's refueling outage A1R13. The inspectors were concerned that operating solid in Mode 4 (RCS temperature from 350 – 200ºF would drastically increase the probability of an LTOP overpressure event and increase the probability of lifting the RH suction and discharge relief valves.

II. <u>INSPECTION ACTIVITIES (IP 71152) -Continued</u>

B.2) Inspection Scope - Continued

- On April 21, 2001, and May 1, 2007, the RH suction relief valve discharged to the Byron HUTs unexpectedly during ECCS check valves testing due to procedure deficiencies (IR 624329). The inspectors were concerned that the corrective actions for these events had not been entered into the licensee's Corrective Action Program.

- It is unclear whether the inputs used in Section 6.2 of calculation CN-CRA-07-50 were verified or received a peer check. In addition, it is unclear whether calculation CN-CRA-08-9 has been approved as a design base calculation.

- Licensee did not provide documentation that licensee staff had reviewed and approved the Westinghouse calculation CN-CRA-08-9 and that the calculation had been entered into the licensee's official system of documents.

- Calculations CN-CRA-07-50 and CN-CRA-08-9 did not address that the HUTs were only designed for 200 degrees F. The calculations ignore the design temperature limit of 200 F for the HUT which is stated in UFSAR section 15.7.2.1. The HUT is designed for only 200 F even though design pressure is 15 psig (29.7 psia). If 350 F steam is discharged to the HUT, the nozzle entering the HUT and the tank around the nozzle will quickly heat up possibly as high as 340 F even with 4.25 inches of water quenching the steam. At 340 F the HUT is 140 F past design. Thermal stresses are very significant on tanks and can cause failures and there is no analysis that indicates any temperature over 200 F is acceptable for the HUT. The condition is worse if the tank has a liner that breaks down at over 200 F.

- Calculations CN-CRA-07-50 and CN-CRA-08-9 did not address the following potential ASME violations. The RH relief valves are designed and allowed by Technical specification to relieve a steam and water discharge at 450 psig and 375 degrees F. The discharge piping was incorrectly routed to the HUTs instead of the PRT as in the Westinghouse design. Contrary to the allowed discharge characteristics, the following conditions exist now and have existed in the past since original construction. The HUTS were supplied under Design Specification F/L-2750. The HUT is a nominal 125,000 gallon vertical tank. The tank shell material is SA 516 Grade *70* with SA 240 Type 304 clad. The nominal shell plate thickness is 0.672. The tanks are designed to ASME Section III, Class 3, 1974 Edition with Summer 1975 Addenda. Licensee's response stated that the common header piping to the HUT was only designed for 30 psig. This was and still is in violation of the ASME code. The EPN numbers for these pipes are: 0ABE2AA, 0AB02AB, 0AB09B, 0AB09CA, 0AB09CB. All of these pipes are 4" schedule 40S.

Note: This list constitutes all the currently known concerns associated with these calculations, but does not preclude additional concerns from being identified during the inspection.

II. <u>INSPECTION ACTIVITIES (IP 71152) -Continued</u>

B.3) The inspectors will document, but not perform a review for the following minor finding identified in allegation file RIII- 2007-A-024 **(Mel has lead).**

The inspectors had previously reviewed licensee calculations and assumptions with respect to gaseous releases from a postulated HUT rupture. Inspectors identified that the licensee analysis did not specifically address how these assumptions bound the actual plant configuration and/or operation of the waste gas systems. The inspectors reviewed the calculation and determined these discrepancies (i.e., quicker release rate, lower initial volume, and cross-connected gas systems) would have an impact on the calculated release rates; however, the margin contained in the calculation was sufficient. The inspectors concluded that the failure to account for actual plant configurations in an accident analysis was a violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control." The inspectors assessed this violation in accordance with IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening," and determined that the finding was minor because the available margin and other conservative assumptions were sufficient to compensate for identified discrepancies. Therefore, in accordance with IMC 0612, this violation of minor significance is not subject to enforcement action in accordance with Section IV of the NRC's Enforcement Policy.

B.4) The inspectors will review the licensee's input assumptions for the expected off-site dose consequences during post LOCA recirculation due to leakage from the ECCS systems. Specifically, the inspectors will review the licensee's basis for apparently excluding dose contributions from scenarios which postulate open or stuck open CVCS, SI and RHR system relief valves during the post LOCA recirculation phase **(Gerry has lead).**

III) <u>INSPECTION CRITERIA</u>

The inspectors will review and evaluate these issues to determine if; (1) the problems were accurately identified; (2) the common causes were adequately ascertained (note that evaluation of root cause is only for significant conditions adverse to quality and may be deferred to the biennial PI&R inspection); (3) extent of condition and generic implications were appropriately addressed; (4) previous occurrences were considered; and (5) corrective actions proposed/implemented were appropriately prioritized to address the problems and were commensurate with the safety significance of the issues. In accordance with IP 71152, it is not expected that the inspectors will assess each attribute for every issue. Rather, inspectors may choose to assess licensee performance against selected attributes, as necessary to be most effective.

IV) <u>COMPLETION STATUS</u>

Because these findings relate to a common issue associated with the design and operation of the HUT, the reviews discussed above will constitute one annual sample of 4-7 annual samples described in Section 02 and Section 05 of IP 71152.

If the inspectors confirm a potential problem with the post LOCA recirculation assumptions used in the off-site dose analysis, the inspectors will document the licensee positions and evaluate licensee corrective actions. If problems with the post LOCA analysis assumptions exist, it will likely become an unresolved issue, which may require further review by staff in NRR (Robert Taylor - Accident Analysis Branch Chief is point of contact).

V) <u>RESOURCE ESTIMATE</u>

IP 71152 Section 3.02(e) "Level of Effort" Identifies that 167 hours for a two Unit Site for the in-depth review of 4-7 issues. Thus, 167/7 = 24 hrs to 167/4 = 41 hrs of review per issue is the expected normal review effort. It is anticipated that the inspectors will likely exceed the upper bounds (41 hrs of on-site inspection) due to the complexity of the issue under review. It is estimated that two inspector weeks (e.g. 32*2 = 64 hrs) will be expended to complete this review scope and overtime will be requested as needed.

VI) <u>REPORT SCHEDULE AND DOCUMENTATION</u>

The inspectors will conduct inspection preparation during the week of September 22 and an information request was E-mailed to the site on September 9th to support inspection preparation. The onsite inspection will occur from October 6-10th and the inspection report documentation will occur the week of October 20th (delay due to engineering seminar the week of October 13th). Mr. O'Dwyer will be responsible for providing the report input for the items identified in Section II.B.2, and II.B.4 above (input due to M. Holmberg by October 24th). Mr. Holmberg has responsibility for providing report input for remaining items in Section II and for overall report preparation. Based upon this documentation schedule, the draft report input will be to the EB1 Chief for approval by October 29th. The report prepared will be a stand alone DRS inspection report (50-454/08-009; 50-455/08-009) and concurrence will include the Plant Support Branch and the EB1 members who performed the reviews and reached conclusions on the findings identified in the Byron Allegation RIII-2007-A-024 file.