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Dominion

April 17, 2007

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

Serial No. 06-140A
KPS/LIC/CDS: R11
Docket No. 50-305
License No. DPR-43

DOMINION ENERGY KEWAUNEE, INC.
KEWAUNEE POWER STATION
RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION REGARDING LICENSE AMENDMENT REQUEST 215, "MODIFICATION OF INTERNAL FLOODING DESIGN BASIS"

Pursuant to 10 CFR 50.90, Dominion Energy Kewaunee, Inc. (DEK) submitted a request for approval of a proposed amendment to the Kewaunee Power Station (KPS) Updated Safety Analysis Report (USAR) (reference 1). The proposed amendment would clarify design criteria associated with internal flooding in the KPS USAR.

Subsequently, the Nuclear Regulatory Commission (NRC) transmitted a request for additional information (RAI) regarding the proposed amendment. The RAI questions and associated DEK responses are provided in attachment 1 to this letter. Attachment 2 provides proposed revisions to the KPS USAR pages included in reference 1. Attachment 3 provides copies of selected references that were used in our responses to the RAI questions.

This RAI response revises the wording of the response to one question in the significant hazards determination discussed in reference 1. However, this revision does not change the conclusions of the no significant hazards determination. The requested approval date for the proposed amendment remains unchanged.

If you have any questions or require additional information, please contact Mr. Craig Sly at (804) 273-2784. A complete copy of this submittal has been transmitted to the State of Wisconsin as required by 10 CFR 50.91(b)(1).

Very truly yours,

Eugene S. Grecheck
Vice President - Nuclear Support Services

Commitments made by this letter: None.

Reference:

1. Letter from Leslie Hartz (DEK) to Document Control Desk, "License Amendment Request 215 – Modification of Internal Flooding Design Basis," dated March 17, 2006.

Attachments:

1. Response to NRC Request for Additional Information Regarding Kewaunee License Amendment Request 215.
2. Proposed Revisions to Marked-up KPS USAR Pages Included in LAR-215.
3. Copies of Selected References.

cc: Regional Administrator, Region III
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COMMONWEALTH OF VIRGINIA)
)
COUNTY OF HENRICO)

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by Eugene S. Grecheck, who is Vice President – Nuclear Support Services of Dominion Energy Kewaunee, Inc. He has affirmed before me that he is duly authorized to execute and file the foregoing document in behalf of that Company, and that the statements in the document are true to the best of his knowledge and belief.

Acknowledged before me this 17th day of April, 2007.

My Commission Expires: August 31, 2008.

Margaret B. Bennett
Notary Public

(SEAL)

ATTACHMENT 1

RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION REGARDING KEWAUNEE LICENSE AMENDMENT REQUEST 215

KEWAUNEE POWER STATION

DOMINION ENERGY KEWAUNEE, INC.

Response to NRC Request for Additional Information Regarding Kewaunee License Amendment Request 215

Pursuant to 10 CFR 50.90, Dominion Energy Kewaunee, Inc. (DEK) submitted a request for approval of a proposed amendment to the Kewaunee Power Station (KPS) Updated Safety Analysis Report (USAR) (reference 1). The proposed amendment would clarify the design criteria associated with internal flooding in the KPS USAR.

Subsequently, on November 17, 2006, the Nuclear Regulatory Commission (NRC) transmitted a request for additional information (RAI) regarding the proposed amendment (reference 11). Each RAI question and DEK's corresponding answer is provided below.

Question 1

The NRC issued Multi-Plant Action Item (MPA) B-11 to track action items of some licensees who did not provide satisfactory responses to an NRC request to certain licensees (plants operating prior to March 1, 1974) to review and determine whether the failure of any non-category I (seismic) equipment could result in a condition that might potentially adversely affect the performance of safety related equipment. Follow-up inspections by the NRC of licensees' completed actions pertaining to MPA B-11 were transferred to the resolution of USI A-17, "Systems Interaction in Nuclear Power Plants". USI A-17 was later resolved as stated in GL 89-18 and the internal flooding portion of USI A-17 was resolved through a satisfactory completion of the actions required by GL 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities-10 CFR 50.54(f)," (IPE).

Your proposed addition to the UFSAR, B.11 INTERNAL FLOODING, refers to MPA B-11 and AEC flooding guidelines. What documents (e.g. correspondence between the NRC and the Kewaunee licensee, or AEC Codes that were in effect at the time of your licensee issuance, or later NRC issued documents) make MPA B-11 and AEC flooding guidelines part of your licensing basis? Please provide copy of, or provide reference to, those document(s).

Response to Question 1

The description of NRC Multi-Plant Action Item (MPA) B-11 stated above is consistent with DEK's understanding of the purpose of MPA B-11. Seven AEC internal flooding guidelines were developed by the AEC staff for use in internal flood mitigation and protection. As stated in the "Introduction" section of attachment 1 to KPS Amendment Request (LAR) - 215 (reference 1):

"...The AEC identified the issue as Multi-Plant Generic Issue B-11, "Susceptibility of Safety Related Systems to Flooding Caused by Failure of Non-Class I Systems." Under this generic issue, the AEC developed a set of guidelines for internal flooding protection. No correspondence has been found indicating these guidelines were sent to Kewaunee for consideration. However, the guidelines have been considered in this proposed change to the Kewaunee internal flooding design basis."

No evidence has been found on the docket or in internal records that either the generic document, MPA B-11, or the subsequent AEC guidelines were sent to Kewaunee by the AEC.

Regarding relevant docketed correspondence, on September 26, 1972 (reference 2), the AEC issued a letter to KPS requesting a review to determine if the failure of any non-Category I (seismic) equipment, particularly in the circulating water and fire protection systems, could result in a condition such as flooding or release of chemicals that might potentially adversely affect the performance of safety-related equipment required for safe shutdown of the facility or to limit the consequences of an accident. This request was prompted by the failure of an expansion bellows in a main condenser circulating water line at Quad Cities Unit 1. The subsequent flooding caused degradation of some safety-related equipment.

The issue of applicability of the Quad Cities incident was discussed before the generic communication of September 26, 1972, as part of the Kewaunee operating license proceedings. As noted in the summary report of the 148th ACRS meeting of August 10-12, 1972, service water system and turbine building flooding was discussed between the AEC staff and the ACRS. It was noted in that summary that, "Staff review of Kewaunee turbine-building drawings failed to disclose a potential for a repetition of the Quad Cities flooding incident of June 9, 1972."

On October 31, 1972, KPS responded to this letter with the results of the requested review (reference 3). The response did not identify any action items or modifications and no additional correspondence was received with regards to internal flooding associated with the Quad Cities operating experience or the September 26, 1972 AEC letter. The AEC issued Supplement No. 1 to the Kewaunee Safety Evaluation on December 18, 1972, and Supplement No. 2 on May 10, 1973. Neither of these supplements addressed or referenced the issue docketed in the two letters noted above. Although the AEC internal flooding guidelines were subsequently sent to and applied to the licensing basis of several plants with design and construction dates similar to Kewaunee, no similar correspondence was sent to Kewaunee.

NUREG-1435, "Status of Safety Issues at Licensed Power Plants," (reference 8) associates MPA B-11 with NRC Temporary Instruction (TI) 2515/88, "Inspection of Licensee's Actions Taken to Implement NRC Guidelines for Protection from Flooding of Equipment Important to Safety" (reference 9). A review of past NRC inspection reports for KPS found no evidence that the NRC conducted an inspection at KPS under TI 2515/88. TI 2515/88 discusses the applicability of the MPA B-11 guidelines and states:

> *"Plants licensed before March 1, 1974 were required to review their facilities and make modifications, as necessary. For plants licensed after that date this issue was addressed as part of the licensing review process. This item is identified by NRR as MPA item B-11 and is applicable to all plants."*

Since the AEC internal flooding guidelines were sent to and applied to the licensing basis of several plants with design and construction contemporaneous to Kewaunee, and lacking any detailed flooding guidance specifically for Kewaunee, the AEC internal flooding guidelines have been conservatively applied in this request to our plant design as well.

Question 2

The new flooding design criteria (f) in reference 1 state, "Protected equipment is sufficient to achieve and maintain safe shutdown requirements." No other functional requirements for protected equipment are given.

(a) USAR B.5 states that Class I items are protected against damage from serious flooding to the extent that the Class I function is impaired. USAR 1.3.1 states that Class I items are vital to safe shutdown and isolation of the reactor and those items whose failure might cause or increase the severity of an accident or result in an uncontrolled release of substantial amounts of radioactivity.

(b) In your letter, reference (B), the criteria for safety equipment function that was not jeopardized by flooding also included the ability "to limit the consequences of an accident."

Please explain why your proposed change to the USAR reduces the functional requirements of Class I items that are protected from internal flooding.

Response to Question 2

General Discussion

The proposed design criterion (f) is not intended to reduce the functional requirements of Class I items. The intent of the proposed design criterion (f) is to establish a criterion for the protection of Class I items required to achieve and maintain "safe shutdown" following an internal flooding event. Therefore, the Class I items that are being addressed by criterion (f) are limited to only those items whose Class I function is required during an internal flooding event.

The term "protected equipment" in the proposed criterion (f) includes any equipment that would not be damaged by flooding, regardless of whether the equipment is protected by a barrier (wall, curb, shield, etc.), is located in an area that will not flood, or otherwise will not fail during an internal flood event. The "protected equipment" required for safe shutdown can include different combinations of plant equipment depending on the evaluated flood scenario. The concept of multiple required sets of plant equipment to achieve safe shutdown is consistent with the design philosophy for the design basis high-energy line break (HELB) outside containment evaluations as documented in KPS USAR section 10A.

The apparent lack of clarity of the intended purpose of criteria (f) warrants consideration for changing the proposed criteria to an actual "functional requirement" statement. The functional requirement for protected equipment following an internal flood event is:

"The plant must withstand the consequences of an internal flooding event in such a manner that it retains the capability to achieve and maintain the reactor in a safe shutdown condition."

In order to properly apply this functional requirement to the proposed flooding design criteria as intended, recommended changes to the proposed USAR section B.11 criteria are presented at the end of this response.

Discussion of RAI Item (a)

The definition of Class I items in USAR Section 1.3.1 is complete and accurate and does not change due to the proposed new flooding design criteria.

USAR section B.5.a requires that Class I items be protected against damage from the rupture of a pipe or tank resulting in serious flooding or excessive steam release to the extent that the Class I function is impaired. USAR section B.5.a does not imply that all Class I items are protected at all times. A "Class I function" during an internal flood event is any function of a Class 1 component directly involved in the mitigation or limitation of the consequences of the flooding event. If a Class 1 component has no function during an internal flood event, then it does not require "protection" during an internal flooding event.

The protection of all Class I equipment or of only needed Class I equipment from damage due to flooding appears to be the focus of this RAI. DEK's position is that only needed equipment be protected based on the original plant licensing basis presented in Section B.5 of the Kewaunee FSAR (reference 5). FSAR Section B.5 contains a discussion of when Class I equipment does not require protection. The KPS FSAR section B.5 states:

"No protection is required if the factors described under a, b, f and g cannot affect any Class I systems, or if redundant systems are provided and the physical separation of these systems is sufficient to prevent these factors from damaging both systems. Under c and d, redundancy and physical separation may decrease the requirements for protection. If redundancy and physical separation are not used, and if the surrounding building is not designed as a missile barrier, missile protection by shielding is necessary, either by shielding the source itself or by shielding the system."

A review of documentation associated with past USAR changes established that an inadvertent administrative error deleted the FSAR paragraph discussed above during the development of Revision 0 of the KPS USAR in 1982. The inadvertent deletion of this paragraph was discovered after the submittal of KPS LAR 215 (reference 1). A change request is planned to re-insert this paragraph into USAR section B.5, so that the USAR will accurately reflect the licensing basis for the protection of Class I items.

The effect of this inadvertent deletion on the USAR changes proposed in LAR 215 is minor since the proposed new design criteria and the flooding evaluations performed to date are not affected. However, the missing paragraph does clarify the basis on which the design criteria and flooding evaluations were performed and why all Class I equipment does not require protection from flooding.

This position is demonstrated by the identification of Class I components needed to respond to a HELB outside containment as described in USAR section 10A. As in the HELB scenario evaluations, the set of needed equipment required to meet the internal flooding design functional requirement of safe shutdown may vary depending on the various flooding scenarios. These needed equipment components are to be protected, but damage may be allowed if redundancy or physical separation of systems allows the plant to maintain the required functional purpose that the damaged component provided. The KPS evaluations for a HELB outside containment are documented in USAR Section 10A, Postulated Failure Analysis, and were accepted by the NRC staff in the KPS Operating License SER, Supplement 2, Section 6.8.3, Safe Shutdown Following the Postulated Pipe Rupture (reference 13).

NRC Task Interface Agreement (TIA) 2005-10 (reference 10) provides some clarification of needed equipment. TIA 2005-10 addresses three issues related to KPS residual heat removal (RHR) pump vulnerability to flooding due to random and seismically induced failures of non-seismic qualified piping. TIA 2005-10 makes the following statement regarding USAR section B.5.a (as currently written in the USAR):

> "USAR, Appendix B, Section B.5, "Protection of Class I Items," states that Class I items are protected against damage from "Rupture of a pipe or tank resulting in serious flooding or excessive steam release to the extent that the Class I function is impaired." Thus, the protection against damage from rupture of a pipe is limited to those conditions where the system is needed to perform its Class I function."

In other words, in the event of a pipe rupture, equipment needed to achieve and maintain safe shutdown should be identified and protected and other equipment (Class I or otherwise) is not addressed or covered for this event because it does not require protection to mitigate the event.

Two additional factors related to USAR section B.5.a are also important to the understanding of the basis behind the functional requirements and design criteria for flooding proposed in LAR 215. First, USAR section B.5.a is the only B.5 item that discusses the extent of damage (i.e., ...to the extent that the Class I function is impaired). Therefore, the Class I "function" is the issue, not Class I "equipment." Secondly, USAR section B.5.a addresses the requirements of both HELB and flooding scenarios together. The concept of "needed equipment" is well documented for HELB, therefore, without any guidance to the contrary, it would appear applicable to internal flooding as well.

Alternate Means are Allowed to Accomplish the Required Function

The HELB evaluations discussed in KPS USAR section 10A and the KPS OL SER (USAR section 6.8.3) also allow for alternative means to accomplish a required function. These documents state that the RHR pumps are not required to achieve safe shutdown since the heat removal function of the auxiliary feedwater (AFW) system and the steam generator (S/G) power operated relief valves (PORVs) can be used to perform the heat removal function. Thus, alternate "systems" are allowed to accomplish the heat removal function. Although the S/G PORVs and the AFW system are not redundant systems of the RHR system, they do provide a redundant function. Similarly, the HELB evaluations accept the use of the safety injection (SI) system pumps as an alternate/redundant means to borate the reactor core during cooldown if the normal chemical volume and control system boration pathways are unavailable due to HELB damage. The use of alternate means to accomplish required functions is a basis used by the AEC in their acceptance of the KPS HELB evaluations. Therefore, the use of alternate equipment to accomplish required functions is allowed when determining the list of needed equipment. The inadvertently deleted FSAR B.5 paragraph directly supports the application of the alternate equipment approach to internal flooding evaluations.

The safe shutdown equipment list (SSEL) developed for internal flooding uses both concepts discussed above. Given the resulting plant conditions determined from flooding evaluations, the SSEL was verified to have sufficient protected equipment to meet the internal flooding functional requirement of achieving and maintaining the reactor in a safe shutdown condition. Therefore, the functional requirements for Class I protected equipment are not reduced since USAR section B.5.a is met and USAR section 1.3.1 is unaffected.

Discussion of Item (b)

Kewaunee's October 31, 1972 letter (reference 3), referenced both the equipment "required for safe shutdown" and equipment "to limit the consequences of an accident" because that was the scope of the requested review. This scope was addressed by assessing the potential consequences of non-category I system failures on "engineered safety systems." The consequences of the three identified lines were stated as acceptable in reference 3 "...because of the safety equipment redundancy and design arrangement, the functional purpose of the safety equipment would not be jeopardized in the event of failure of any of these lines." This basis for acceptability is consistent with USAR section B.5.a requirements for protection of equipment. However, the requested review did go beyond the requirements of USAR B.5.a for individual event scenarios such as internal flooding, since the evaluation considered, as requested, ESF equipment such as containment spray pumps which are needed for a LOCA but not internal flooding events.

The concept of protecting only the equipment needed to achieve and maintain safe shutdown is discussed extensively in previous paragraphs, but it is also supported by enclosure 3 to Temporary Instruction (TI) 2515/88, "Inspection of Licensee's Actions Taken to Implement NRC Guidelines for Protection from Flooding of Equipment Important to Safety" (reference 9), which states:

> "No single incident of a Non-class I system component or pipe failure shall prevent the safe shutdown of the facility."

No mention is made in TI 2515/88 of equipment required to limit the consequences of an accident beyond those required for safe shutdown.

When every subset of needed equipment for every design basis event is compiled together, the resulting integrated set represents all Class I functions and components. In general terms, that compiling of subsets is the intent of section B.5 in both the FSAR and USAR. However, when the individual contributing events are considered separately, only the subset required for safe shutdown from that event is necessary to meet the licensing mandate to protect the public. The proposed USAR section B.11 is intended only to address internal flooding events, therefore only those subsets of Class I functions and components necessary to address safe shutdown under flooding events are required. Not all Class I components require protection from flooding during an internal flooding event.

This position is also consistent with the NRC procedure for verifying the seismic adequacy of equipment which was provided to the industry as an enclosure to Generic Letter 87-02 (reference 14). The procedure provides assumptions regarding the scope of seismic adequacy reviews as follows:

1. The seismic event does not cause a loss-of-coolant accident (LOCA, a steam line break accident (SLBA), or a high energy line break (HELB), and a LOCA, SLBA, or HELB does not occur simultaneously with or during a seismic event. However, the effects of transients that may result from ground shaking should be considered.
2. Offsite power may be lost during or following a seismic event.
3. The plant must be capable of being brought to a safe shutdown condition following a design-basis seismic event.

Change to Proposed New USAR Section B.11

In order to address any confusion regarding the intent of criteria (f) and to avoid the potential for future misinterpretation, DEK proposes:

1. Removal of the current criteria (f), and
2. Changing the first sentence of the proposed Section B.11.2 to the following:

   "The plant must withstand the consequences of an internal flooding event in such a manner that it retains the capability to achieve and maintain the reactor in a safe shutdown condition. Toward this end, the design criteria for internal flooding evaluations are:"

Changes to the supporting text of proposed new USAR Section B.11 are also required to reflect this criteria revision. See attachment 2 of this letter for a complete copy of the proposed USAR Section B.11 revisions.

Question 3

In your submittal, Reference (A), Attachment 1, under Section 2, PROPOSED CHANGE, item d. you state that that an additional single failure is not considered.

In view of the Criterion on page B.5-1 of the USAR which states:

> *No single event will cause failure of redundant circuits or Engineered Safety Feature components in a manner such that a single failure after the event could prevent the protective functions of the associated Engineered Safety Features.*

In what specific systems are you proposing to not consider the additional failure?

Response to Question 3

The proposed addition of USAR section B.11 does not alter the design basis criteria for an additional single failure for any plant systems. The requirements of the Engineered Safety Features are unchanged. The current licensing basis does not require the consideration of an additional failure for seismic or tornado events which represent the most likely potential initiators of an internal flood scenario. Flooding as a consequence of a HELB outside containment does consider an additional single failure.

With regard to non-HELB internal flooding events, the KPS position that "...an additional single failure is not considered" is based on three items from the original licensing basis for KPS. First, section 1.3 of the KPS USAR states that the plant was:

> "...designed, constructed and is being operated to comply with Wisconsin Public Service Corporation's (WPSC) understanding of the intent of the AEC General Design Criteria (GDC) for Nuclear Power Plant Construction Permits, as proposed on July 10, 1967."

The USAR description of the 1967 AEC GDCs, as applied to the plant design, comprises part of the KPS licensing basis.

KPS's USAR Section 1.8, Criterion 2, "Performance Standards," states the following:

> *"Those systems and components of reactor facilities which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of their consequences shall be designed, fabricated, and erected to performance standards that will enable the facility to withstand without loss of the capability to protect the public. The additional forces that might be imposed by natural phenomena such as earthquakes, tornadoes, flooding conditions, winds, ice and other local site effects. The design bases so established shall reflect:*

(a) *appropriate consideration of the most severe of these natural phenomena that have been recorded for the site and the surrounding area, and*

(b) *an appropriate margin for withstanding forces greater than those recorded to reflect uncertainties about the historical data and their suitability as a basis for design."*

Although USAR section 1.8, Criterion 2 is related to natural phenomenon, if an internal flooding event were to occur, it would most likely be the result of either an earthquake, tornado, or HELB event. The earthquake and tornado are addressed in Criterion 2 above. There is no consideration or requirement of an additional single failure specified in the design criterion for these events and absent such a requirement, there is no need to invoke such a requirement to ensure protection of the public during such natural events. As previously stated, the HELB outside containment event, as a design basis accident, requires consideration of an additional single failure and that event is separately and appropriately considered.

Second, during the license review process, KPS responded to the AEC September 26, 1972 letter regarding the Quad Cities event (reference 2) by letter dated on October 31, 1972 (reference 3). In our response we identified several Non-Category I (seismic) lines and dispositioned their failure by stating that:

> "...However, because of safety equipment redundancy and design arrangement, the functional purpose of the safety equipment would not be jeopardized in the event of failure of any of these lines."

With regard to this conclusion about the failure of a non-seismic line, this statement explicitly notes that it is acceptable to lose a piece of safety equipment because the functional purpose can still be accomplished by a redundant or physically separated safety equipment component. Therefore, the criterion for an additional single failure does not apply. This is reiterated in TIA 2005-10 (reference 10) in the response to TIA Question 1.

Third, as further confirmation regarding the application of an additional single failure to other section B.5 events, such as fire or missiles, an in-house review of the KPS USAR and the KPS OL SER was performed. The review did not identify any non-HELB event requirements for single failure criterion to be considered other than those addressed by GDC Criterion 41, which is discussed below.

Additional support for the position that an additional single failure does not apply to flooding events also comes from the MPA B-11 guidelines (reference 4). These guidelines specify the assumption of a coincident loss of offsite power in flood evaluations, but do not specify any assumption regarding an additional single failure.

The RAI specifically identifies the Criterion on page B.5-1 as a basis for the application of an additional single failure to flooding, however, the applicability of this Criterion is limited to HELB evaluations only as discussed below.

During the review of original KPS FSAR section B.5 (reference 5) for the response to RAI #2, it was discovered that the "No single event..." criterion statement regarding an additional single failure was not included in the original FSAR and that it (or similar wording) did not exist elsewhere in the original FSAR. Meetings between Kewaunee, Prairie Island, and AEC personnel in September 1972 resulted in acceptance of the AEC-DRL position on this subject (reference 12). Both Kewaunee and Prairie Island agreed to make the necessary plant modifications to comply. The AEC-DRL position was related to protection of the reactor protection system and engineered safety features from high energy pipe ruptures, jet impingement, or pipe whip reactions and applies to USAR section B.5.b only. The requirement was a result of HELB outside containment concerns and appears to be the precursor statement to item 20 of a December 15, 1972 letter from the AEC (reference 15). In the early 1980s, this statement was added (appropriately) to the KPS USAR, Revision 0. This statement was, however, inappropriately placed within KPS USAR, Revision 0, section B.5. As currently positioned in USAR section B.5, the "No single event..." statement appears applicable to all section B.5 criteria. The Criterion is, however, only applicable to the HELB criteria of section B.5.b.

Per the guidance of the December 15, 1972 AEC letter on HELB (reference 15), an additional single failure is assumed in the HELB outside containment evaluations for KPS and is assumed in the evaluation of the "needed equipment" listings provided in USAR section 10A. Since the HELB outside containment has the potential to initiate a reactor cooldown event, engineered safety features may be required to respond. Accordingly, protection against an additional single active failure is the appropriate accident response strategy and typical accident response practice. Consequential flooding resulting from a HELB (submergence) is also considered in the availability of needed equipment.

A USAR change request is planned to properly associate the "No single event..." statement exclusively to events involving high energy pipe ruptures, jet impingement, or pipe whip reactions. Associating the "No single event..." statement exclusively to HELB scenarios in USAR section B.5 is consistent with the licensing basis for single failure consideration stated in USAR Section 1.8, Criterion 41:

> "<u>Criterion 41 – Engineered Safety Features Performance Capability</u>
>
> Engineered safety features such as emergency core cooling and containment heat removal systems shall provide sufficient performance capability to accommodate partial loss of installed capacity and still fulfill the required safety function. As a

minimum, each engineered safety feature shall provide this required safety function assuming a failure of a single active component.

Answer

Sufficient redundancy and duplication is incorporated into the design of the engineered safety features to insure that they may perform their function adequately even with the loss of a single active component. Details of the capability of these systems under normal and component malfunction conditions are included in Sections 6 and 9. An analysis of the adequacy of these systems to perform their functions is included in Section 14."

Therefore, the proposed addition of USAR section B.11 does not alter the design basis criteria for an additional single failure for any plant systems and the requirements of the Engineered Safety Features are unchanged. The proposed criteria is intended to clearly state that the evaluations of protected equipment for non-HELB flood events does not require the consideration of an additional single failure.

QUESTION 4

The new flooding design criteria in reference (A), states in part, "c. Pipe and tank failures assume the single most limiting failure in an area, as determined by maximum flood level calculated in that area."

While it is true that a single failure of a non-category I SSC is to be considered for each particular internal flooding evaluation, a break in a non-category I SSC, that delivers the maximum flood level in an area, should not eliminate evaluating a smaller break in the same area, which could cause greater damage to safety equipment. For example, a small diameter pipe break near safety related switchgear may cause a more limiting failure due to spraying, than a larger diameter pipe break in the same area away from the switchgear.

Please explain how the effects of water spraying on safety related equipment is considered in your flooding design criteria.

Response to Question 4

The effects of spray from a break in a non-category I SSC were considered in the original flood evaluations summarized in reference 3. Specifically, the potential for spray from lines over the safety injection pumps and the containment spray pumps was identified. Both sets of pump motors were confirmed to have drip-proof enclosures and are protected from spray. Additionally, one Motor Control Center (MCC) associated with engineered safety features was identified as potentially vulnerable to spray. Although the MCC may be lost, because of redundancy, safety functions supported by this MCC are not jeopardized.

The effects of spray from a break in a non-category I SSC were also addressed in the more recent KPS flooding analyses using a different approach. Rather than evaluating the case-by-case geometry of potential flood sources and necessary safe shutdown equipment, a zone approach was used. A Safe Shutdown Equipment List (SSEL) for flooding was created by modifying the existing Appendix R Fire SSEL in two ways. First, equipment included in the fire SSEL solely due to concerns of spurious system operation was removed. Second, some additional instrumentation was added for operational convenience. This approach specifies more equipment than is actually needed in the flood SSEL and is considered to be conservative.

The effects of fire or flooding in an area are similar regarding the status of equipment in an area. During a fire all equipment in the respective fire zone is assumed to be unavailable during the fire and for achieving hot shutdown. In some cases, post-fire manual operation of valves is necessary to achieve cold shutdown (reference App R fire description). The fire SSEL demonstrates there is sufficient equipment outside the fire zone to safely shut down the plant. Application of this concept to flooding assumes unavailability of all the equipment within the fire zone boundary containing the flood source while maintaining the ability to safely shut down the plant using only equipment outside the zone.

The designated fire zones at Kewaunee are larger than the designated zones for the flooding evaluations, thereby bounding the set of equipment assumed to be unavailable after an internal flooding event. Water flowing from a pipe rupture is not restricted to the originating flood zone and may proceed beyond individual flood zone boundaries. However, the effects of spray are limited to the originating flood zone. Water collection from spray would flow as pathways allowed, and is bounded by the maximum flood water evaluations which address both transient and steady-state flood levels. It is likely that spray would only affect equipment in a localized area within an individual flood zone. However, the zone approach conservatively assumes failure of all equipment in the flood zone containing the flood source.

Therefore, the flood SSEL conservatively bounds the effects of spray. Class I components may be lost to flood waters or to spray, but such equipment is either not on the flood SSEL or its loss is acceptable because of redundancy (i.e. the safety function is not jeopardized).

Change to Proposed New USAR Section B.11

Although spray has been addressed in the zone approach used to demonstrate the ability to achieve safe shutdown, it was not specifically listed as a design criterion. In the response to RAI No. 2 above, criterion (f) was deleted and addressed with new verbiage in a new location. The following new criterion (f) is proposed.

> "f. The effects of water spraying, dripping, or splashing on sensitive equipment from pipes and tanks not capable of maintaining their pressure boundary during a seismic event are to be considered in the assessment of available equipment. "

Changes to the supporting text of proposed USAR Section B.11 are also required to reflect this proposed new criterion (f). See Attachment 2 of this letter for a complete copy of the proposed USAR Section B.11 revisions.

<u>Question 5</u>

Reference (A), Attachment 1 page 5, states "For internal flooding, safe shutdown is defined as hot shutdown."

Please provide the reference to the licensing document that establishes the basis for the above statement, i.e. safe shutdown for internal flooding at Kewaunee is hot shutdown?

<u>Response to Question 5</u>

In a Safety Evaluation Report regarding low temperature overpressure protection systems at KPS (reference 19), the NRC stated:

> "The Kewaunee plant was licensed on the basis that reaching hot shutdown status is sufficient."

There are no licensing documents specifically docketed to Kewaunee that establish a safe shutdown definition for internal flooding other than those associated with HELB related flooding. Part of the intent of the proposed amendment was to formally specify the safe shutdown condition for internal flooding events as hot shutdown. The proposed definition of flooding safe shutdown as hot shutdown is based on the following:

A definition of safe shutdown for internal flooding is provided in NUREG-1174, "Evaluation of Systems Interactions in Nuclear Power Plants, Technical Findings Related to Unresolved Safety Issue A-17" (reference 16). The NUREG contains an appendix entitled, "Internal Flooding and Water Intrusion Insights," which states the following:

> "Safe-shutdown equipment for a flood or water intrusion event would typically include the equipment to perform the following functions:
>
> - Bring the plant to hot shutdown and establish heat removal.
> - Maintain support systems necessary to establish and maintain hot shutdown.
> - Maintain control room functions and instrumentation and controls necessary to monitor hot shutdown."

Safe shutdown definitions for other events do appear in the Kewaunee docketed licensing basis. The USAR Section 10A.2.4, "Criteria for Plant Operability Following Pipe Rupture," includes criteria for plant operability to maintain the ability to bring the reactor to a hot shutdown condition, and to assure that the reactor can eventually be brought to a cold shutdown condition. The Kewaunee OL SER, Supplement 2 (reference 13) notes that, in general, the equipment used to maintain hot shutdown after a HELB can be utilized to achieve cold shutdown, although this approach is not the normal cooldown method. The OL SER credits the hot shutdown equipment for cooldown and removal of decay heat as well as boron injection to maintain adequate shutdown margin. Both internal flooding and HELB are the subject events of USAR section B.5.a.

The NRC SER (reference 18) associated with the Kewaunee response to USI A-46 (GL 87-02, reference 14) states that the licensee should be able to bring the plant to, and maintain, a hot shutdown condition during the first 72 hours following a safe shutdown earthquake. Failure of a non-seismic pipe or tank following a seismic event may initiate an internal flooding event.

Therefore, the basis for the proposed definition of safe shutdown in LAR 215 is NUREG-1174 (reference 16), which defines "safe shutdown" as hot shutdown for flood or water intrusion events. In addition, the proposed definition is consistent with the docketed Kewaunee licensing basis for related design basis events.

<u>Question 6</u>

Reference (A), section 3, states that a corrective action program (CAP) was issued concerning the peer review of Kewaunee's PRA for IPE. The CAP identified eight issues associated with internal flooding.

Please provide a summary of those eight issues and their resolutions.

<u>Response to Question 6</u>

In June of 2002, a peer review of the Kewaunee PRA model, which was used for the December 1992 IPE response (reference 17), was performed by the Westinghouse Owners Group. Eight issues were identified during the peer review. Each issue was entered into the Kewaunee corrective action program. A summary of each issue and its resolution is provided below. All of the resolutions were accomplished through updates/revisions to the Kewaunee flooding PRA model.

<u>Issue 1</u>: Pipe failures resulting in rupture were excluded from the analysis. Only leaks were considered credible.

<u>Resolution</u>: Pipe ruptures are now explicitly modeled using EPRI TR-1012302, "*Pipe Rupture Frequencies for Internal Flooding Probabilistic Risk Assessments (PRAs).*"

<u>Issue 2</u>: Water propagation through doors with gaps less than 1/8-inch were ignored without regard to the ability to stop continued leakage.

<u>Resolution</u>: The PRA flooding model was revised to assume doors stop flow initially, but in the long term, fluid levels equalize on both sides.

<u>Issue 3</u>: Back-flow through drains was considered but was stopped once the flooding source was isolated. It was not clear how continued back-flow would be stopped until fluid levels equalize between connected rooms.

<u>Resolution</u>: The PRA flooding model was revised to consider the effects of back-flow through drains until fluid levels equalize.

<u>Issue 4</u>: Operator action to terminate flooding events was assumed to occur at an estimated time with essentially 100% success.

<u>Resolution</u>: The Human Reliability Analysis (HRA) methodology in the PRA flood model has been revised to utilize the Human Cognitive Reliability/Operator Reliability Experiment (HCR/ORE) model for flooding diagnosis and the Technique for Human Error Rate Prediction (THERP) model for isolation.

Issue 5: Human action dependencies between the flooding and mitigation action were not addressed.

Resolution: Local mitigation actions considered in the flooding PRA model are now evaluated to determine if the flooding scenario would impair the operator's ability to mitigate (identify and isolate) a flood source.

Issue 6: The potential to cause flooding through maintenance and testing or special system configurations was not considered.

Resolution: The PRA model has been revised such that maintenance-induced failures are explicitly addressed and included as part of the initiating event frequency.

Issue 7: There was no evidence of a search of plant-specific initiating events that might be relative to flooding.

Resolution: During the evaluation of maintenance activities, a search was conducted to identify maintenance that could cause a flood (e.g., manway opening without isolating line, equipment repairs). None were found.

Issue 8: Flood frequencies were based on very old generic data.

Resolution: The flood frequency data used in the IPE PRA flood model has been replaced with data from EPRI TR-1012302, Revision 1 (released March of 2006).

Question 7

*Reference (A), Attachment 1, Regulatory Safety Analysis, section 5.1.1 states, "The proposed change to incorporate design criteria into the USAR provides added administrative assurance that internal flooding will be appropriately addressed, consistent with existing functional requirements, **and that safety related SSCs will not be affected by a potential failure of a non-safety related SSC**." (bold type added).*

The statement in bold type above does not agree with what is stated in reference (B), which says that failure of non-category I (seismic) systems could potentially adversely affect the performance of engineered safety systems. Reference (B) seemingly states a less stringent licensing position than your more current statements in reference (A). ***Please explain.***

Response to Question 7

The NRC staff's observation that the specified statement in Section 5.1.1 of reference (A) is inconsistent with reference (B) is correct. The reference (B) statement accurately reflects the assessment of the potential failure of non-seismic systems as documented during the operating license review period for Kewaunee. The reference (A) section 5.1.1 statement was intended to state that the incorporation of the proposed design criteria would provide added assurance that plant safety (including the safety-related SSCs that are needed for safe shutdown) will not be affected by a potential failure of a non-safety related SSC.

Change to Proposed New USAR Section B.11

This proposed statement does not enhance the overall understanding of the significant hazard consideration question being addressed. Therefore, the first full paragraph on page 9 of Attachment 1 in Reference (A) should be revised as follows:

> "The proposed change to incorporate design criteria into the USAR provides added administrative assurance that internal flooding will be appropriately addressed, consistent with existing functional requirements for safety-related SSCs. The change does not affect any accident initiators or the facility accident analysis. Thus, the probability and the consequences of an accident remain unchanged."

Question 8

The new flooding design criteria in reference (A) does not consider Class I pipe to fail.*

Is Class I* pipe designed to maintain its pressure boundary during a seismic event?

Response to Question 8

Yes, Class I* pipe is designed to maintain its pressure boundary during a seismic event. Section B.2.1.a of the Kewaunee USAR provides the following definition of Nuclear Safety Design Classification for Class I and Class I* components:

> "Those structures and components including instruments and controls whose failure might cause or increase the severity of a loss-of-coolant accident or result in an uncontrolled release of substantial amounts of radioactivity, and those structures and components vital to safe shutdown and isolation of the reactor. Some items in Table B.2-1 are designated as Class I* indicating that these items have been designed to Class I Design Basis Earthquake loading (dynamic) only, and that these items are treated as Class III items in all other respects."

USAR Table B.7-1 provides the load combinations for components by Class of component. Table B.7-1 shows Class I and Class I* to be identical. Therefore, piping designed as Class I* is designed to maintain its pressure boundary during a design basis seismic event.

Change to Proposed New USAR Section B.11

The RAI addresses this criterion from a "functional" perspective (i.e. pressure boundary). The proposed B.11.2 criterion (a) states, "Only non-Class I/I* pipe or tanks are considered to fail..." The criterion also states that "...individual items may be determined not to fail if evaluated to withstand the Design Basis Earthquake (DBE)." Since several non-Class I/I* piping segments at Kewaunee have been qualified to withstand a DBE without loss of pressure boundary, two items are evident. First, the pipe class identification is not a stand-alone criterion. Second, the function to, "...maintain its pressure boundary during a seismic event," is a definitive criterion.

Proposed criterion (a) should be revised to reflect the generic functional requirement of pressure boundary and that, for clarity, dependence on the actual pipe class is not specified. The revised Section B.11.2.a item should read as follows:

"a. Only pipes and tanks not capable of maintaining their pressure boundary during a seismic event are considered to fail."

Changes to the supporting text of proposed USAR Section B.11 are also required revision to reflect this proposed criterion revision. See Attachment 2 of this letter for a complete copy of the proposed USAR Section B.11 revisions.

Inherent to this revised criteria are the elements from the originally proposed criteria a). That is, that pipes and tanks that are seismically designed or have been evaluated to withstand a seismic event do not fail and, therefore, are not considered as sources of internal flooding. Class I and Class I* components are not considered as internal flood sources since they have been designed and installed to withstand a DBE. In addition, the Class II and Class III* piping have been designed and installed to withstand an OBE as documented in KPS USAR Section B.4.5 and Table B.7-1. Therefore, these pipes are also not considered to be potential flooding sources.

NRC TIA 2001-02 dated August 29, 2002 (reference 20) was written for Prairie Island. Issue No. 2 in the TIA specifically discusses the seismic qualification associated with the UBC Zone 1 criteria. Prairie Island and Kewaunee were designed by the same Architect Engineering Firm (Pioneer), licensed to the same design basis, and built at the same time with the same piping installation standards. The NRC response to TIA 2001-02 (Issue 2) states that piping designed to the UBC Zone 1 loadings are essentially designed for the Operational Basis Earthquake (OBE). This equivalence is clearly applicable for KPS as demonstrated in USAR Table B.7-1 which identifies the UBC loads for the OBE condition of loading. The OL SER's for both plants support this statement.

TIA 2001-02 concludes that, "...it is consistent with the plant's licensing basis to use UBC Zone 1 loadings to show that non-Class I SSCs will not adversely affect Class I SSCs during a design-basis event." It is apparent that this conclusion was a factor in the selection of piping systems evaluated by Pioneer in the preparation of the October 31, 1972 response (reference 3) to the Quad Cities flooding event. Only piping that was designated as Class III was considered a potential internal flood source around ESF equipment. This conclusion is consistent with the position that Class II and Class III* components were designed not to fail during the worst seismic event (OBE) anticipated to occur during the lifetime of the plant.

Although the USAR defined distinction of Class III is that it did not require additional UBC Zone 1 loads, research into the installation of Class III piping has shown that Class III piping was installed no differently than the Class II or Class III* piping. All Class II, III, and III* piping installed in the seismic qualified areas of the plant were installed to the requirements of USAS B31.1.0-1967 with applicable N-code cases to ASA B31.1-1955 per the Pioneer piping design and installation specifications. Seismic loading and acceleration factors which met or exceed the criteria for an OBE (UBC Zone 1) seismic event were stated in these specifications. Required QC inspections verified that piping of any class was installed to its specifications. To be clear, Kewaunee does not contend that Class II, III, and III* piping is seismically qualified. It is our belief, however, that Class III piping, like Class II and Class III* piping is capable of withstanding an OBE and, "...will not adversely affect Class I SSCs during a design-basis event." Therefore, Class II, III, and III* piping do not represent potential internal flooding sources.

As a result of recent internal flooding concerns, many non-Class I pipes have been specifically evaluated by walkdowns and analysis to verify their status following a seismic event. The evaluated pipes include both Class II and Class III piping systems. All pipes evaluated to date have confirmed that the piping segments will remain intact and maintain their pressure boundary during a DBE, which represents loadings that are twice the required OBE per TIA 2001-02 and the KPS OL SER. Accordingly, it is Kewaunee's position that all piping classes defined in USAR Section B.2.1 would maintain their pressure boundary during the worst anticipated seismic event, designated as the OBE, which is the criteria that was deterministically used during original licensing to address this issue.

There are some water distribution lines in the facility that are not installed to the standards of the Class II, III, or III* piping system specifications. These pipes are considered as potential internal flooding sources and can only be excluded as flooding sources following specific structural evaluations confirming acceptability or by conducting appropriate modifications.

References:

I. References from NRC Request for Additional Information Questions

(A) Letter from Leslie Hartz (DEK) to Document Control Desk, "License Amendment Request 215 – Modification of Internal Flooding Design Basis," dated March 17, 2006.
(B) Letter from E.W. James (WPSC) to R.C. DeYoung (NRC), "WPS Review of Non-Category I (Seismic Equipment)," dated October 31, 1972.

II. References from Request for Additional Information Responses

** Indicates that this reference is provided in Attachment 3 to this document.

1. Letter from Leslie Hartz (DEK) to Document Control Desk, "License Amendment Request 215 – Modification of Internal Flooding Design Basis," dated March 17, 2006.
2. Letter from R. C. DeYoung (AEC) to E. W. James (WPSC), dated September 26, 1972.
3. Letter from E. W. James (WPSC) to R.C. DeYoung (NRC), "WPS Review of Non-Category I (Seismic Equipment)," dated October 31, 1972.
4. Multi-Plant Generic Issue B-11, "Susceptibility of Safety Related Systems to Flooding Caused by Failure of Non-Class 1 Systems."
5. Letter from C. W. Geisler (WPPS) to H. R. Denton (NRC), "Updated Final Safety Analysis Report," dated July 21, 1982. (see Section B.5)**
6. Generic Letter 89-18, "Resolution of Unresolved Safety Issue A-17, Systems Interactions in Nuclear Power Plants," dated September 6, 1989.
7. Generic Letter 88-20, "Individual Plant Examinations for Severe Accident Vulnerabilities – 10 CFR 50.54 (f)," dated November 23, 1988.
8. NUREG-1435, Supplement 4, "Status of Safety Issues at Licensed Power Plants," published December 1994.
9. NRC Temporary Instruction 2515/88, "Inspection of Licensee Actions Taken to Implement NRC Guidelines for Protection from Flooding of Equipment Important to Safety," dated April 6, 1987. **
10. Memorandum from E. M. Hackett (NRC) to M. A. Satorius (NRC), "Final Response to Task Interface Agreement 2005-10 Relating to Impact of Flooding on Residual Heat Removal (RHR) Pumps at Kewaunee Power Station (Task Interface Agreement (TIA) 2005-10)(TAC NO. MC8937)," dated May 5, 2006. **
11. E-mail from D. Jaffe (NRC) to T. Breene (KPS) dated November 17, 2006.
12. Pioneer Engineering & Services Memorandum/Telephone Log, Protection of Reactor Protection System and Engineered Safety Features from Pipe Rupture, Jet Impingement, or Pipe Whip Reactions," dated September 25, 1972. **

13. "Supplement No. 2 to the Safety Evaluation by the Directorate of Licensing U. S. Atomic Energy Commission in the Matter of Wisconsin Public Service Corporation, Wisconsin Power and Light Company and Madison Gas and Electric Company, Kewaunee Nuclear Plant, Kewaunee County, Wisconsin, Docket No. 50-305," dated May 10, 1973.
14. Generic Letter 87-02, "Verification of Adequacy of Mechanical and Electrical Equipment in Operating Reactors, Unresolved Safety Issue (USI) A-46," dated February 19, 1987.
15. Letter from A. Giambusso (AEC) to E. W. James (WPS), "General Information Required for Consideration of Effects of a Piping System Break Outside Containment," dated December 15, 1972.
16. NUREG-1174, "Evaluation of Systems Interactions in Nuclear Power Plants, Technical Findings Related to Unresolved Safely Issue A-17," dated May 1, 1989. **
17. Letter from C. R. Steinhardt (WPSC) to NRC, "Response to Generic Letter 88-20, Individual Plant Examination," dated December 1, 1992.
18. Letter from W. O. Long (NRC) to M. L. Marchi (WPSC), "Safety Evaluation Report for USI A-46 Program Implementation (TAC No. M69455)," dated April 14, 1998.
19. Letter from S. A. Varga (NRC) to D. C. Hintz (WPSC), dated September 6, 1985. **
20. Memorandum from L. B. Marsh (NRC) to J. A. Grobe (NRC), "Response to Task Interface Agreement (TIA 2001-02) and Task Interface Agreement (TIA 2001-04) Regarding Evaluation of Service Water System Design Basis Requirements at Prairie Island (TAC NOS. MB1402, MB1403, MB1855, and MB1856)," dated August 29, 2002.

ATTACHMENT 2

RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION REGARDING LICENSE AMENDMENT REQUEST 215, "MODIFICATION OF INTERNAL FLOODING DESIGN BASIS"

PROPOSED REVISIONS TO MARKED-UP KEWAUNEE USAR PAGES INCLUDED IN LAR 215

KEWAUNEE POWER STATION

DOMINION ENERGY KEWAUNEE, INC.

B.11 INTERNAL FLOODING

B.11.1 GENERAL DISCUSSION

Internal flooding can occur as a result a rupture of a pipe or tank in a system containing or connected to a large volume of water. This section does not address flooding from other liquids such as chemicals or diesel fuel that are stored in tanks. In these cases, cubicles or dikes contain liquids due to f~~F~~ailure of ~~these~~ non-seismic components ~~are contained by cubicles or dykes~~ or spillage occurs remote from any safety-related equipment.

Internal flooding resulting from sources outside containment (other than natural phenomenon) was addressed in the original licensing process for Kewaunee. Amendment 17 to the FSAR addressed internal flooding from a postulated rupture in a service water line in the vicinity of the diesel generator rooms. The postulated rupture of a high-energy line (HELB) that also includes flooding consequences was addressed by FSAR Amendment Nos. 24, 27, and ~~28 which~~28 that added Appendix 10A to the FSAR. Appendix 10A provided detailed design criteria and assessments of potential HELB events. Although the rupture of a service water pipe was addressed in the FSAR, the general criteria for the evaluation of internal flooding from a rupture of a pipe or tank was not captured in the FSAR.

In 2005, re-constitution of the design criteria for internal flooding was initiated in support of several internal flood protection modifications. When the operating license for Kewaunee was issued, the AEC was pursuing the issue of internal flooding for previously licensed plants via the Multi-Plant Generic Issue B-11, "Susceptibility of Safety Related Systems to Flooding Caused by the Failure of Non-Class I Systems." The AEC developed a set of guidelines for internal flooding protection. These guidelines were not sent to Kewaunee for consideration; however, the guidelines have been considered in the re-constitution of the internal flooding design criteria.

This section applies only to internal flooding resulting from the ~~rupture~~ failure of a pipe or tank below the criteria for high-energy systems as a result of a seismic event. The HELB design criteria is addressed specifically in Section 10A.

B.11.2 FLOODING DESIGN CRITERIA

The plant must withstand the consequences of an internal flooding event in such a manner that it retains the capability to achieve and maintain the reactor in a safe shutdown condition. Toward this end, the design criteria for internal flooding evaluations are:~~The design criteria for internal flooding evaluations are:~~

(a) Only pipe and tanks not capable of maintaining their pressure boundary during a seismic event are considered to fail.~~;~~

(b) Only failures in piping and branch runs exceeding 1 inch are considered.

(c) Pipe and tank failures assume the single most limiting failure in an area as determined by maximum flood level calculated in an area.

(d) Operator actions and design features are considered, but an additional single failure is not.

(e) Flooding is assumed coincident with the loss of offsite power if it increases the consequences of a flood.

(f) The effects of water spraying, dripping, or splashing from non-seismically qualified sources on sensitive equipment are to be considered in the assessment of available equipment.

Safe shutdown following an internal flood is defined as hot shutdown. The reactor can be maintained in the hot shutdown condition for an extended period of time, if necessary, for cold shutdown equipment repairs.

~~Some non-~~Class I/I* pipes and tanks are excluded from consideration due to their seismic qualification. Other pipes have been excluded from consideration as a flood source based on seismic evaluations to verify that the pipes would have reasonable assurance to sustain the combined effects of a design basis earthquake and deadweight loading without a loss of pressure boundary function. This assurance is obtained from experience based evaluations and/or by bounding evaluations. Criteria from the ASME Section III Code for evaluation for level D loading or from ASME Section III Appendix F can be used to establish reasonable assurance against leakage from a pressure boundary.

~~The consideration of only non-Class I/I* (or seismically evaluated) pipe or tank failures is consistent with the original AEC flooding guidance developed as part of Multi-Plant Generic Issue B-11. Likewise, the assumption that only one such failure is assumed is also consistent with AEC flooding guidelines.~~ The failure of a pipe or tank is assumed as a result of DBE seismic loads. Only one pipe or tank component is assumed to fail. The failure is conservatively assumed to be the worst case (complete double-ended rupture) with respect to flooding potential in each area evaluated. The consequences of lesser breaks resulting in dripping or spray are also considered. Multiple pipe or tank failures are not considered in the analysis for a pipe or tank rupture because the potential interactions, such as pipe whip or jet impingement, are not applicable for lines that are not defined as high-energy lines. As discussed in Section 10A, high-energy lines would consider additional failures as a consequence of the initial rupture, if warranted. Multiple failures resulting from seismic loadings are also not considered as credible because of the robust design of non-Class I/I* piping. Specific evaluations of non-Class I/I* piping in the Class I portion of the Turbine Building basement (Safeguards Alley) and the Auxiliary Building have demonstrated that the Class II and Class III piping in ~~this~~ these areas ~~area~~ are capable of withstanding the effects of a DBE without failure. The piping in ~~this area~~ these areas was installed to the same standards ~~as~~ used throughout the seismically qualified areas of the station ~~as a whole and, therefore, is typical of all station piping. However, it is not possible to ensure that all non-Class I/I* tanks and piping would remain intact during a DBE without additional evaluation. Accordingly, a non-Class I/I* pipe or tank that is not seismically evaluated is assumed to fail as a result of DBE seismic loads. Only one pipe or tank component is assumed to fail. The failure is conservatively assumed to be the worst case (complete double-ended rupture) with respect to flooding potential in each area evaluated.~~

Operator actions and design features are considered in the evaluation of internal flooding consequences. The design features include level sensing devices to alert operators to take action, check valves to prevent backflow through pipes, barriers to protect safety-related equipment (including existing walls, doors, dikes, etc.), and circulating water pump trips to minimize flood sources. Operator actions in response to control room indications are the primary means of identification and termination of flooding sources.

Loss of offsite power (LOOP) is assumed unless the LOOP results in less limiting consequence. Design features that rely on electric power to operate (such as sump pumps) are only credited for flood protection if they are powered by site emergency power sources.

~~Safe shutdown following an internal flood is defined as hot shutdown. The reactor can be maintained in the hot shutdown condition for an extended period of time, if necessary, for cold shutdown equipment repairs.~~

B.11.3 CLASS I EQUIPMENT PROTECTION

The criteria for Class I equipment protection is stated in Section B.5.a. It states that Class I items are protected against damage from the rupture of a pipe or tank resulting in serious flooding to the extent that the Class I function is impaired. Consistent with the AEC flooding guidelines, the Class I functions required following the rupture of a pipe or tank which results in internal flooding are those functions necessary to achieve and maintain safe shutdown of the reactor. For internal flooding, safe shutdown is defined as hot shutdown. The ability to achieve and maintain safe shutdown demonstrates the effectiveness of the plant design and flood protection measures to protect necessary Class I equipment.

The installed flood protection measures include drain line check valves, flooding barriers, level alarms, and a circulating water pump trip. These measures provide additional protection to the original plant design against flood damage. The criteria for protection of Class I equipment has not changed, however, the means by which to comply with the criteria has become more effective.

The AEC flooding guidelines developed in response to Multi-Plant Generic Issue B-11 do not specify that flood protection equipment ~~needs~~ is to be safety related. Flood protection equipment is not intended to mitigate any aspect of a design basis accident. Therefore, consistent with the Kewaunee quality classification criteria, such equipment does not meet the criteria to be classified as safety related.

B.11.4 CONCLUSION

The ability to cope with internal flooding from the rupture of a pipe or tank is determined per the criteria provided in B.11.2 above. Equipment required for the safe shutdown of the reactor must be protected from the flood consequences consistent with Section B.5.a.

REFERENCES - APPENDIX B

1. Morris, Hansen, Holley, Biggs, Namyet, and Minami, "Structural Design for Dynamic Loads", McGraw-Hill Co., Inc., New York, 1959

2. RA Wiesemann, RE Tome, R. Salvatori, "Ultimate Strength Criteria to Ensure No Loss of Function of Piping and Vessels Under Earthquake Loading", WCAP 5890 Rev 1, October 1967.

3. George W. Housner, "Vibration of Structures Induced by Seismic Waves", Shock and Vibration Handbook, Volume III, McGraw-Hill, Inc., New York, 1961

4. E. L. Vogeding, "Topical Report, Seismic Testing of Electrical and Control Equipment", WCAP 7817, December 1971

5. "Report Covering the Effects of a High Pressure Turbine Rotor Fracture and Low Pressure Turbine Disc Fractures at Design Overspeed", Westinghouse LTD, Report B, E & M.

6. RC Gwaltney, "Missile Generation and Protection in Light-Water-Cooled Power Reactor Plants", ORNL-NSIC-22, September 1968.

7. J. N. Fox, "Likelihood and Consequences of Turbine Overspeed at the Point Beach Nuclear Plant", WCAP 7525, August 1970

8. John A. Blume & Associates, Engineers, "Kewaunee Nuclear Power Plant-Earthquake Analysis of the Reactor-Auxiliary-Turbine Building, JAB-PS-01, February 16, 1971" (submitted as part of Amendment No. 9 to this license application)

9. John A. Blume & Associates, Engineers, "Kewaunee Nuclear Power Plant-Earthquake Analysis: Reactor-Auxiliary-Turbine Building Response Acceleration Spectra", JAB-PS-03, February 16, 1971 (submitted as Amendment No. 9 to this license application)

10. "Methodology for Calculating the Probability of a Missile Generation from Rupture of a Low Pressure Turbine Disc" - Revision 1, dated July 1980, Westinghouse

11. "Results of Probability Analyses of Disc Rupture and Missile Generation"-Revision 0, dated August 1980, Westinghouse

12. "WPS Kewaunee Missile Probabilities/Probability of Disc Rupture and Missile Generation Due to Stress Corrosion" - Letter to DC Hintz from Philip E. Mescher dated August 9, 1982 (Letter #PM-229-82)

13. NRC Safety Evaluation Report - Letter to ER Mathews from SA Varga dated October 26, 1981 (K-81-174)

REFERENCES – APPENDIX B (cont'd)

14. Supplement No. 1 to Generic Letter (GL) 87-02 which transmits Supplemental Safety Evaluation Report No. 2 (SSER No. 2) on SQUG Generic Implementation Procedure, Revision 2 as corrected on February 14, 1992 (GIP-2), dated May 22, 1992

15. Letter from H. L. Thompson (NRC) to Licensees, Letter No. K-85-132 dated June 28, 1985

16. NRC SER, SA Varga (NRC) to CW Giesler (WPS), Letter No. K-84-61 dated March 16, 1984

17. Letter from C. R. Steinhardt (WPSC) to the NRC Document Control Desk, dated September 17, 1992

18. Letter from C. R. Steinhardt (WPSC) to the NRC Document Control Desk, dated February 18, 1993

19. Seismic Qualification Utility Group (SQUG), "Generic Implementation Procedure (GIP) for Seismic Verification of Nuclear Power Plant Equipment", Revision 2 as corrected February 14, 1992

20. Letter from C. R. Steinhardt (WPSC) to the NRC Document Control Desk, Letter No. NRC-89-56, dated May 05, 1989, "Criteria for Determining Continued Operability of Safety Related Piping Systems"

21. Letter from C. R. Steinhardt (WPSC) to the NRC Document Control Desk, Letter No. NRC-96-016, dated February 13, 1996, "Response to Generic Letter 95-07"

22. Letter from C. R. Steinhardt (WPSC) to the NRC Document Control Desk, Letter No. NRC-96-071 dated July 18, 1996, "Response to Request for Additional Information - Generic Letter 95-07"

23. NRC Safety Evaluation Report of Licensee Response to Generic Letter 95-07, □.Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves", Letter to ML Marchi (WPSC) from R. J. Laufer (NRC), dated January 13, 1998 (K-98-008)

24. Letter from WO Long (NRC) to ML Marchi (WPSC), "Kewaunee Nuclear Power Plant-Safety Evaluation Report for USI A-46 Program Implementation", Letter No. K-98-47, dated April 14, 1998

25. Seismic Qualification Utility Group (SQUG), "Generic Implementation Procedure (GIP) for Seismic Verification of Nuclear Power Plant Equipment", Revision 3, May 16, 1997

26. Supplemental Safety Evaluation Report No. 3 (SSER No. 3) on the Review of Revision 3 to the Generic Implementation Procedure for Seismic Verification of Nuclear Power Plant Equipment updated May 16, 1977, (GIP-3), (TAC No. M93624)

REFERENCES – APPENDIX B (cont'd)

27. FSAR Amendment 17 dated May 12, 1972 from E. W. James (WPS) to P. A. Morris (AEC).

28. FSAR Amendment 24 dated January 24, 1973 from E. W. James (WPS) to J. F. O'Leary (AEC).

29. FSAR Amendment 27 dated March 16, 1973 from E. W. James (WPS) to J. F. O'Leary (AEC).

30. FSAR Amendment 28 dated April 13, 1973 from E. W. James (WPS) to J. F. O'Leary (AEC).

31. "Safety Evaluation of Kewaunee Nuclear Power Plant, Supplement 2" dated July 24, 1972.

32. NUREG-0800, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Reactors (LWR Edition) dated July 1981.

33. Letter October 31, 1972 to R. C. DeYoung (NRC) from E. W. James (WPS).

34. Letter dated October 31, 1974 from the AEC to Surry Power Station. This letter contains the guidelines for the evaluation of internal flooding based on MPA B011 (Multi-Plant Generic Issue B-11).

ATTACHMENT 3

RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION REGARDING LICENSE AMENDMENT REQUEST 215, "MODIFICATION OF INTERNAL FLOODING DESIGN BASIS"

COPIES OF SELECTED REFERENCES

REFERENCE PROVIDED IN THIS ATTACHMENT

KEWAUNEE POWER STATION

DOMINION ENERGY KEWAUNEE, INC.

List of References included in this Attachment

1. Not Included.
2. Not Included.
3. Not Included.
4. Not Included.
5. Letter from C. W. Geisler (WPPS) to H. R. Denton (NRC), "Updated Final Safety Analysis Report," dated July 21, 1982 (Section B.5)
6. Not Included.
7. Not Included.
8. Not Included.
9. NRC Temporary Instruction 2515/88, "Inspection of Licensee Actions Taken to Implement NRC Guidelines for Protection from Flooding of Equipment Important to Safety," dated April 6, 1987.
10. Memorandum from E. M. Hackett (NRC) to M. A. Satorius (NRC), "Final Response to Task Interface Agreement 2005-10 Relating to Impact of Flooding on Residual Heat Removal (RHR) Pumps at Kewaunee Power Station (Task Interface Agreement (TIA) 2005-10)(TAC NO. MC8937)," dated May 5, 2006.
11. Not Included.
12. Pioneer Engineering & Services Memorandum/Telephone Log, Protection of Reactor Protection System and Engineered Safety Features from Pipe Rupture, Jet Impingement, or Pipe Whip Reactions," dated September 25, 1972.
13. Not Included.
14. Not Included.
15. Not Included.
16. Attachment Only - NUREG-1174, "Evaluation of Systems Interactions in Nuclear Power Plants, Technical Findings Related to Unresolved Safely Issue A-17," dated May 1, 1989.
17. Not Included.
18. Not Included.
19. Letter from S. A. Varga (NRC) to D. C. Hintz (WPSC), dated September 6, 1985.
20. Not included.

Reference 5

Letter from C. W. Geisler (WPPS) to H. R. Denton (NRC), "Updated Final Safety Analysis Report," dated July 21, 1982 (Section 5.B)

B.5 PROTECTION OF CLASS I ITEMS

Criterion: No single event will cause failure of redundant circuits or Engineered Safety Feature components in a manner such that a single failure after the event could prevent the protective functions of the associated Engineered Safety Features.

The Class I items are protected against damage from:

a. Rupture of a pipe or tank resulting in serious flooding or excessive steam release to the extent that the Class I function is impaired.

b. Pipe whip and steam/water jets following a pipe rupture of an adjacent pipe.

c. Earthquake, by having the ability to sustain seismic accelerations adopted for purposes of plant design without loss of function. Protection from interaction with the surrounding buildings is accomplished by providing a separating joint of sufficient size for earthquake displacements. Unless the building is designed to Class I seismic design, an analysis is made to demonstrate that it will not collapse; otherwise, the systems are protected locally.

d. Tornado wind loads.

e. Other natural hazards. Examples of these hazards are seiche and ice.

f. Fire, in such a way that fire and operation of fire-fighting equipment does not cause damage to redundant parts of the system.

g. Missiles from different sources. These sources comprise:

   (i) Tornado created missiles.

   (ii) Missiles from components containing moving parts, which could be subjected to overspeed. (Potential sources for such missiles are turbines, turbine generators, and diesel engines, gas turbines).

   (iii) Missiles from high-pressure steam and feedwater piping. (These missiles are limited to non-back-seated valve stems and parts bolted to valves with bolts smaller than 3".)

Reference 9

NRC Temporary Instruction 2515/88, "Inspection of Licensee Actions Taken to Implement NRC Guidelines for Protection from Flooding of Equipment Important to Safety," dated April 6, 1987

TEMPORARY INSTRUCTION 2515/88

INSPECTION OF LICENSEE'S ACTIONS TAKEN TO IMPLEMENT NRC GUIDELINES FOR PROTECTION FROM FLOODING OF EQUIPMENT IMPORTANT TO SAFETY

2515/88-01 PURPOSE

To verify that equipment important to safety will not be damaged by flooding caused by the rupture of a non-Class I system component or pipe to the extent that engineered safety features will not perform their design functions. This temporary instruction (TI) is one of a series that describes NRC inspection requirements and guidance needed to verify satisfactory completion of licensee actions in response to multi-plant action (MPA item B-11).

2515/88-02 OBJECTIVES

To compare the actions of the licensee with the 1972 NRC guidelines (Enclosures 1-3) for protection from flooding of equipment important to safety.

2515/88-03 RESPONSIBILITIES AND AUTHORITIES

03.01 Associate Director for Inspection and Technical Assessment, NRR

a. Coordinate with the regional offices to obtain specific information and identify items to be inspected.

b. Coordinate with regions as required to complete the requirements of this TI.

c. When requested by the regions, conduct a review of the results of the inspections. Determine whether further generic action needs to be taken after the completion of the effort directed by this TI.

03.02 Region Management

a. Coordinate with NRR as needed to perform the inspection requirements of this TI.

b. At most facilities, this inspection effort is expected to verify satisfactory licensee implementation of the requirements imposed under this MPA. In such cases, the regional offices will be able to verify satisfactory completion and report MPA verification as complete. However, if the inspection produce results which are unclear and require additional technical resolution, the regional offices should contact the Associate Director for Inspection and Technical Assessment, NRR for resolution. Recommendations for additional action should be provided, if appropriate.

2515/88-04 BACKGROUND

A technical issue was identified concerning the failure of non-safety grade equipment such as condenser bellows which could lead to flooding of safety-related equipment and loss of safety functions. Plants licensed before March 1, 1974 were required to review their facilities and make modifications, as necessary. For plants licensed after that date this issue was addressed as part of the licensing review process. This item is identified by NRR as MPA item B-11 and is applicable to all plants.

Modifications such as water tight doors, curbs, and changes to floor drains may be needed for some plants to solve potential flooding problems. All modifications are plant specific.

2515/88-05 BASIC REQUIREMENTS

For each of the categories listed below, perform examinations in areas susceptible to flooding or water impingement. This may be accomplished by visual inspections or reviews of engineering drawings, operating procedures, and surveillance records. Some examples of equipment and components whose functions may be affected by flooding are motor control centers, electrical switchgear, batteries, diesel generators, and pump and valve controls. Areas susceptible to flooding or water impingement may be adjacent to water supplies for fire suppression, general service and cooling.

05.01 Separation for Redundancy. Determine that redundant equipment important to safety is separated and protected to ensure operability in the event a non-Class I system or component fails and causes flooding or water impingement.

05.02 Access Doors and Alarms. Determine that the watertight barriers for protection from flooding of equipment important to safety have access doors and hatches fitted with switches that annunciate in the control room when the access is open. The access doors should be watertight and functional. Determine the date of the most recent verification of the seal integrity, including a check for watertightness.

05.03 Sealed Water Passages. Determine that penetrations through walls of rooms containing equipment important to safety are sealed against water leakage from a failure of non-Class I water systems. Determine that openings between floors do not create a potential for flooding. For example, there may be unsealed pipe sections in the horizontal structure.

05.04 Floor Drains and Curbs. Determine that floor drains are not obstructed, that screen covers are in place, and that curbs are continuous. Assure that the area served by the floor drains are free from objects which may migrate during water flow to the drain and obstruct the drain (such as poly bags or sheets, paper, etc). Where applicable, determine that floor drain check valves open and close correctly without sticking.

05.05 Water Level Alarms and Trips. Determine that the level alarms and pump trips in rooms containing non-Class I system components and pipes whose rupture could result in flood damage to equipment important to safety alarm in the control room and limit flooding. Determine the date of the most recent verification of the functioning and calibration of the level alarms. Redundance of switches is required; critical trip circuits should be redundant.

05.06 Equipment Location and Protection. Determine that Class I equipment is located or protected such that rupture of a non-Class I system that is a portion of a pumped system or is connected to a cooling tower containing water will not result in failure of the equipment from flooding. For example, water impingement shielding may be used for electric motors.

05.07 Loss of Offsite Power. Determine that the simultaneous loss of off-site power with the rupture of a non-Class I system component or pipe will not affect the operation of the annunciators, alarms, switches, trips, etc.

05.08 Integrity of Class 1E Electrical Systems. Determine that enclosures in harsh environments including high-energy line breaks that contain Class 1E electrical terminals and terminations are sealed/gasketed to prevent moisture intrusion which may cause power system shorting or inoperability of the instrument or component served by the Class 1E source.

05.09 Administrative Control. It is important to maintain protection from flooding of equipment and the licensee may have established administrative control programs to ensure that measures taken in each of the above categories are effective and include periodic surveillances to verify the adequate continuation of such measures. Determine the methods or programs used to keep this protection effective and current. For example, it may be found in portions of repair, maintenance, and inspection procedures or it may be part of design reviews.

2515/88-06 REPORTING REQUIREMENTS

06.01 Regional inspection results shall be transmitted to the Associate Director for Inspection and Technical Assessment, NRR. The inspection effort shall be documented in a routine inspection report.

06.02 Some or all of the inspection requirements of this TI may have been previously accomplished as part of inspections conducted at a particular facility. In such cases where the basis for findings resulting from these inspection requirements is adequately documented in an earlier inspection report, enter the inspection report number, completion date, and other pertinent data in the SIMS data base for the affected facility.

06.03 This TI may serve as a substitute for the applicable portions of the following inspection procedures (IP):

a. IP 62702 Maintenance Program
b. IP 62703 Monthly Maintenance Observations

06.04 When inspection activities required by this TI are completed, enter the status of these activities in the following SIMS data fields. The SIMS issue number for this TI is MPA-B-11.

a. Inspection Report Number. Up to five inspection report numbers may be entered to identify those instances where the inspection activities are documented in more than one inspection report.

b. Inspection Report Date. This data field lists either the date of the final inspection report on this item, the date of the most recent inspection report on this item, or a projected final inspection date for this item.

c. Comments. This data field contains 300 characters and can be used to describe the status of NRC inspection activities for this item at each plant. Useful information in this field would include mentioning of outstanding open items or future licensee action needed to close the item, if applicable.

2515/88-07 EXPIRATION

The TI shall remain in effect until April 1, 1988.

2515/88-08 CONTACT

Questions regarding this TI should be addressed to Paul Cortland, (301) 492-4175.

2515/88-09 STATISTICAL DATA REPORTING

Record actual time spent to perform the inspection and the time spent on followup items identified in the inspection report against module number 25588.

END

Enclosure

Reference 10

Memorandum from E. M. Hackett (NRC) to M. A. Satorius (NRC), "Final Response to Task Interface Agreement 2005-10 Relating to Impact of Flooding on Residual Heat Removal (RHR) Pumps at Kewaunee Power Station (Task Interface Agreement (TIA) 2005-10) (TAC NO. MC8937)," dated May 5, 2006

May 5, 2006

MEMORANDUM TO: Mark A. Satorius, Director
Division of Reactor Projects
Region III

FROM: Edwin M. Hackett, Deputy Director **/RA/**
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

SUBJECT: FINAL RESPONSE TO TASK INTERFACE AGREEMENT 2005-10 RELATING TO IMPACT OF FLOODING ON RESIDUAL HEAT REMOVAL (RHR) PUMPS AT KEWAUNEE POWER STATION (TASK INTERFACE AGREEMENT (TIA) 2005-10)(TAC NO. MC8937)

By memorandum dated November 17, 2005, Region III submitted Task Interface Agreement (TIA) 2005-10, which requested assistance from the Office of Nuclear Reactor Regulation (NRR) to resolve the following three issues related to RHR pump vulnerability to flooding which may result due to seismically induced or random failures of non-seismically qualified piping at Kewaunee Power Station (Kewaunee):

- Does Kewaunee's licensing basis require the RHR system to be protected from seismically induced or random flooding to maintain its at-power operable status in accordance with Technical Specifications?

- Does Kewaunee's licensing basis require the RHR system to be protected from seismically induced or random flooding to maintain its below-hot-shutdown operable status in accordance with Technical Specifications?

- Is RHR operability mode specific? If the RHR system is inoperable below hot shutdown, is the system inoperable above hot shutdown? If it is determined that the RHR system is operable above hot shutdown but inoperable below hot shutdown, what would be the required licensee action?

By memorandum dated April 5, 2006, NRR issued a draft TIA response prepared by NRR's Division of Safety Systems, Balance-of-Plant Branch which Region III was requested to review and provide its comments to NRR's Division of Operating Reactor Licensing within 30 days. On April 21, 2006, NRR held a telephone conference with Region III to discuss the draft TIA response. By memorandum dated April 26, 2006, Region III provided comments on the draft TIA response. This final TIA reflects Region III's comments.

Docket No. 50-305

Enclosure:
NRR Staff Assessment

CONTACT: David H. Jaffe, NRR/DORL
(301) 415-1439

MEMORANDUM TO: Mark A. Satorius, Director
Division of Reactor Projects
Region III

FROM: Edwin M. Hackett, Deputy Director ***/RA/***
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

SUBJECT: FINAL RESPONSE TO TASK INTERFACE AGREEMENT 2005-10 RELATING TO IMPACT OF FLOODING ON RESIDUAL HEAT REMOVAL (RHR) PUMPS AT KEWAUNEE POWER STATION (TASK INTERFACE AGREEMENT (TIA) 2005-10)(TAC NO. MC8937)

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Docket No. 50-305
Enclosure:
NRR Staff Assessment

CONTACT: David H. Jaffe, NRR/DORL
(301) 415-1439

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FINAL STAFF ASSESSMENT

OFFICE OF NUCLEAR REACTOR REGULATION

IMPACT OF FLOODING ON RESIDUAL HEAT REMOVAL PUMPS AT THE KEWAUNEE POWER STATION

TASK INTERFACE AGREEMENT 2005-10

1.0 INTRODUCTION

The Kewaunee Power Station (Kewaunee) licensee (currently Dominion Energy Kewaunee, Inc.) completed an internal flooding analysis in June 2005, as part of an extent of condition review for the station's response to flooding in the auxiliary building under the licensee's calculation 2005-05708, "Internal Flood Levels Due to Postulated Piping Ruptures In General Pipe Lines In Auxiliary Building" Revision 1. Region III inspectors noted that this evaluation identified several potential flooding sources, including a non-seismically mounted service water pipe and non-seismically mounted condensate line, which could result in both residual heat removal (RHR) pump pits being filled with water to over 8 feet and cause both RHR pump motors to fail.

Given that both pumps were vulnerable to flooding from a single-failure, the inspectors questioned the operability of the RHR system in the emergency core cooling system mode. The licensee asserted that this condition was known and is acceptable because the plant is a "hot shutdown" plant and no credit for RHR is given in reaching and maintaining hot shutdown, in accordance with the licensee's response to unresolved safety issue (USI) A-46, following a seismic event. The licensee also stated that a loss-of-coolant accident (LOCA) is not assumed coincident with a seismic event or flood. The inspectors also questioned the operability of RHR in the decay heat removal mode. The licensee asserted that flooding events were not part of the licensing basis for RHR which is based on hot shutdown conditions and concluded that both RHR trains were operable. The initial questions by the residents were characterized as an Unresolved Item in the 2nd quarter integrated inspection report (05000305/2005008). No performance deficiency was identified pending resolution of the issue.

2.0 BACKGROUND

The issues identified by the inspectors involve the definition of operability. Technical guidance provided in Part 9900 of the Nuclear Regulatory Commission (NRC) Inspection Manual lists the following principal criteria for technical specification operability requirements that are relevant to the issue:

> The system operability requirements that are based on safety analysis of specific design-basis events for one mode or condition of operation may not be the same for all modes or conditions of operation.

ENCLOSURE

> The system operability requirements extend to necessary support systems regardless of the existence or absence of support system requirements. The operability of necessary support systems includes regulatory requirements. It does not include consideration of the occurrence of multiple (simultaneous) design-basis events.

Section 1.3.1, "Overall Plant Requirements," of the Kewaunee Updated Safety Analysis Report (USAR) states that those systems and components vital to safe shutdown and isolation of the reactor or whose failure might cause or increase the severity of an accident or result in an uncontrolled release of substantial amounts of radioactivity are designated Class I. In Appendix B to the USAR, Table B.2-1 lists the RHR system as a Class I System. USAR, Appendix B, Section B.5, "Protection of Class I Items," states that Class I items are protected against damage from "Rupture of a pipe or tank resulting in serious flooding or excessive steam release to the extent that the Class I function is impaired." Thus, the protection against damage from rupture of a pipe is limited to those conditions where the system is needed to perform its Class I function. The need for the subsequent performance of the Class I function is informed by earlier licensing basis documents related to pipe rupture or failure.

Section 9.3.3, "System Evaluation," of the Kewaunee USAR describes specific design features protecting the RHR pumps from flooding. The design features include separate, shielded compartments with floor drains. The floor drains direct water to the RHR pump pit sump, where two 60 gpm sump pumps are provided to pump collected water to the waste holdup tank or the deaerated drains tank. Each drain line has a remotely operated valve that automatically closes on high level within the RHR pump compartment, which would indicate either massive failure or the inability of the sump pumps to handle the leakage. The RHR pump pit sump has a high level alarm which will cause an alarm in the main control room on high water level. Each line from the containment sump to the respective RHR pump suction has two remotely operated isolation valves to isolate RHR following a failure of an RHR pump seal or minor pipe break within the RHR pump room. These features provide protection during performance of the Class I function of the RHR system to provide long-term post-LOCA recirculation cooling.

The earliest documents pertaining to seismically-induced or random pipe failure and subsequent flooding were developed in 1972, following the failure of a circulating water system expansion joint at Quad Cities. These documents were concerned with the failure of non-seismic piping systems and the potential flooding of equipment needed for safe shutdown. By letter dated September 26, 1972, the NRC requested Wisconsin Public Service Corporation (a previous Kewaunee licensee) to review Kewaunee to determine whether the failure of any non-Category I (seismic) equipment could result in a condition, such as flooding, that might adversely affect the performance of safety-related equipment required for safe shutdown of the facility or to limit the consequences of an accident. In its response dated October 31, 1972, the licensee stated that the failure of reactor makeup water and demineralized water lines in the auxiliary building basement could potentially adversely affect the performance of engineered safety systems. However, the licensee also stated that, because of safety system redundancy and design arrangement, the functional purpose of the safety equipment would not be jeopardized. The criteria used to make this assessment were not documented.

By letter dated November 7, 1972, the licensee responded to an oral NRC staff request to address random pipe breaks in systems containing high-energy fluids. Sections I through III of the enclosure to that letter provided analyses of postulated breaks in the main steam and main feedwater piping within the auxiliary building, and Section IV of that enclosure described analyses of miscellaneous piping systems. The analyses of miscellaneous piping systems included evaluations of potential flooding effects from failures of the service water, component cooling, demineralized water, and reactor makeup water systems. For these evaluations, the licensee determined that either the system has too low a volume to endanger engineered safety features or the rate of rise of water level was low enough to allow operator action before affecting safeguards equipment. Again, the specific criteria used in these assessments, such as the break size and the operator response time, were not documented.

In their comments on the draft TIA response, Region III staff asked whether a September 23, 1971, letter to Wisconsin Public Service Corporation from the staff was part of the staff review of the TIA and whether the contents of the letter are relevant to the questions in the TIA. This historical reference contained a question (Kewaunee Final Safety Analysis Report (FSAR) Question 8.16) regarding the potential failure of service water piping in proximity to the emergency diesel generator rooms. This question requested that the licensee provide an analysis of the effect of a rupture of one of the service water lines on the emergency power systems. The response to FSAR Question 8.16 was included with Amendment No. 13 to the Application for Construction Permit and Operating License for the Kewaunee Nuclear Power Plant, issued December 15, 1971. The response stated that the rupture of a service water pipe in an emergency diesel generator room could result in loss of the generator or safeguards electrical bus in that room. In addition, the response stated that operation of service water valves from the control room would isolate the break and, if required, operators would realign service water supplies through intact piping. This response is consistent with the licensing basis defined in the later letters dated October 31 and November 7, 1972, in that design arrangement would limit the immediate effects and operator action would limit the later effects such that the functional capability of essential safety equipment would be retained. However, the emergency power system is an essential system in achieving hot shutdown, and the staff questioned the potential failure of the seismically qualified service water lines in the vicinity of the emergency diesel generators based on their safety significance. The later documents establish the licensing basis with respect to pipe failures elsewhere.

To further assess protection from pipe breaks in high-energy systems, the NRC issued a letter to Wisconsin Public Service Corporation dated December 15, 1972. This letter was generic in the sense that identical requests for information were sent to all plants operating or under construction, and it is commonly referred to as the Giambusso letter. The review criteria included with the Giambusso letter are available as an attachment to Standard Review Plan Section 3.6.1. The Giambusso letter only addressed failure of high-energy piping systems; flooding concerns associated with moderate-energy piping systems such as service water were not within the scope of the requested review.

The final analyses of postulated pipe breaks in high-energy systems were largely described in Amendment No. 24 to the Kewaunee Final Safety Analysis Report, with additional information provided in Amendment Nos. 25, 27, and 28. The NRC staff evaluation of these analyses were documented in Supplement 2 to the Licensing Safety Evaluation Report for the Kewaunee Nuclear Power Plant, dated May 10, 1973. The NRC staff's evaluation of pipe breaks outside of containment did not encompass the flooding issues of concern in the TIA because the evaluation focus was on pipe breaks in high-energy systems. The NRC staff's evaluation clearly stated that the RHR system was not essential to bring the plant to a safe, cold shutdown, in the event of a pipe break in a high-energy system. However, the NRC staff noted that the equipment used to achieve cold shutdown under normal plant operating conditions, which includes the RHR system, would also be used following a high-energy pipe rupture, if the equipment were still available.

Two generic safety issues were also relevant to the licensing basis for protection against internal flooding. These issues were USI A-17, "Systems Interactions in Nuclear Power Plants," and USI A-46, "Seismic Qualification of Equipment in Operating Nuclear Power Plants." The resolution of USI A-17, as presented in Generic Letter (GL) 89-18, expected licensees to evaluate the potential for water intrusion and flooding from internal sources in the Individual Plant Examination (IPE) process requested by GL 88-20. The Kewaunee IPE did not identify any significant risk associated with internal flooding, and, therefore, no modifications addressing internal flooding vulnerabilities were implemented. The resolution of USI A-46 involved the verification of seismic adequacy of mechanical and electrical equipment in nuclear power plants, as documented in GL 87-02. The scope of the seismic verification was limited to components whose failure could damage equipment necessary to maintain the plant in a safe shutdown condition for 72 hours. This criterion excludes protection of the RHR system cold shutdown functions from the scope of seismic verification.

3.0 BALANCE-OF-PLANT Branch (SBPB) RESPONSE TO REGION III REQUESTS

Region III requested the Office of Nuclear Reactor Regulation to respond to three questions. The SBPB response is summarized below.

Question 1:

> Does Kewaunee's licensing basis require the RHR system to be protected from seismically induced or random flooding to maintain its at-power operable status in accordance with Technical Specifications?

SBPB Response:

The licensing basis of the RHR system in the power operation, startup, and hot standby modes of operation includes protection of the RHR system function from failures of non-seismically qualified piping, consistent with the licensee's statements in the letter dated October 31, 1972. In addition, the RHR system function in these operating modes is required to be protected from a single random failure of an RHR pump seal or an unspecified minor piping failure within the RHR pump pits during the long-term post-accident recirculation phase. The licensing basis includes a general statement that operator action or the limited system volume would preclude failure of engineered safety features as a result of other random pipe breaks, but this statement does not imply that the RHR system is protected from these other random pipe failures.

Operability of the RHR system at Kewaunee in the power operation, startup, and hot standby modes of operation is based on the RHR system performing its emergency core cooling and long-term post-accident recirculation cooling functions. These functions are clearly specified as Class I functions of the system, and the RHR system must be capable of performing these functions assuming a failure of non-seismic piping. However, the licensing basis includes a statement that the design arrangement and redundancy of the safety systems ensure the system function can be performed following failure of a non-seismic pipe. Therefore, the design basis of the RHR system must include a provision that the trains be separated in a manner that prevents simultaneous damage to both trains from a failure of a non-seismic pipe. However, protection of both trains from any failure of non-seismic piping is not necessary to satisfy the licensing basis.

The design basis for the RHR system includes ensuring that a failure of the RHR pump seal or other unspecified passive failure in one loop of the RHR system during the long-term, post-LOCA recirculation mode of operation would not result in failure of the opposite RHR loop. Section 9.3.3 of the Kewaunee USAR identifies specific flood protection features for the RHR pump pit rooms. These features include the drain lines from each room to the sump, the automatic isolation of the drain line on high level in the associated pump room, the operation of two sump pumps to transfer collected water from the sump to the liquid radioactive waste system, the capability to remotely isolate individual lines from the containment sump to the RHR pump suction to limit the inventory of water that can leak into a room, and alarms in the main control room actuated by high sump or high pump room levels. Consistent with the criteria for operability, the ability of these flood protection features to perform their design functions is necessary to maintain the RHR system operable. These features may also be credited when evaluating the potential for failure of a non-seismic pipe to simultaneously affect both RHR trains such that the system function is lost.

Although the features described above provide some protection against other random failures of piping that may lead to flooding of the RHR pump rooms, the capability to mitigate these types of events was not included in the design basis of those features. The operability of the RHR system in these operating modes is based on LOCA mitigation, and the NRC staff has generally concluded a coincident random failure of seismically qualified piping was sufficiently unlikely that it could be excluded from the design basis of emergency core cooling system components. Therefore, the potential for random piping failures to overwhelm these protective features and flood the RHR pump rooms does not affect the at-power operable status of the RHR system. The plant is safe because alternate systems can perform the RHR function necessary for safe shutdown.

In their comments on the draft TIA response, Region III staff suggested that the licensing basis requires that the RHR trains be protected in a manner which ensures that the Class 1 function is maintained for both trains. The NRR staff agrees that this criteria applies to systems necessary to achieve safe shutdown following a pipe or tank failure that occurs as an initiating event, such as the auxiliary feedwater system and the emergency power system, because these systems must perform their function following an additional single failure as described in Section B.5 of the Kewaunee USAR. However, in licensing the Kewaunee plant, the staff specifically accepted exclusion of the RHR system from the set of systems necessary to achieve safe shutdown following such an event, which effectively means the RHR system has no Class I function following a random break in a seismically qualified pipe.

The licensing basis, as documented in Section IV of the attachment to the licensee's letter dated November 7, 1972, includes the statement that the rate of rise of water level from any piping failure would be low enough to allow operator action before affecting safeguards equipment. Conformance with this licensing basis capability may involve development of alarm response procedures for alarm conditions associated with pipe failures (e.g., the RHR pump room high level alarm) to ensure operator actions are completed quickly enough to avoid an adverse affect on safeguards equipment.

In summary, the answer to the first question is "yes" with respect to failure of non-seismic piping to the extent the failure threatens the RHR system function (protection against flooding that could result in loss of a single train is not required) and "no" with respect to random pipe breaks in the auxiliary building since the Kewaunee RHR system function is only required to be protected from a single random failure of an RHR pump seal or an unspecified minor piping failure within the RHR pump pits during the long-term post-accident recirculation phase.

Question 2:

> Does Kewaunee's licensing basis require the RHR system to be protected from seismically-induced or random flooding to maintain its below-hot-shutdown operable status in accordance with Technical Specifications?

SBPB Response:

Similar to the licensing basis of the RHR system in the power operation, startup, and hot standby modes of operation, the Kewaunee licensing basis includes protection of the RHR system shutdown cooling function from failures of non-seismically qualified piping, consistent with the licensee's statements in the letter dated October 31, 1972. The Kewaunee licensing basis for the RHR system in the shutdown cooling mode of operation includes no discussion regarding protection from the effects of random, moderate-energy pipe breaks.

Wisconsin Public Service Corporation stated in the letter dated October 31, 1972, that the functional purpose of safety equipment would not be jeopardized by the failure of non-seismic piping in the auxiliary building because of safety equipment redundancy and design arrangement. In the cold shutdown and refueling modes of operation, the RHR system performs an essential safety function that cannot readily be performed by other systems or components. Therefore, the determination of RHR system operability in the cold shutdown and refueling modes of operation involves an assessment of whether the failure of non-seismic piping would result in failure of both RHR trains. In completing this assessment, the licensing basis allows the consideration of design features that separate the RHR system trains.

In summary, the answer to the second question is "yes" with respect to failure of non-seismic piping to the extent the failure threatens the RHR system function (protection against flooding that could result in loss of a single train is not required) and "no" with respect to random pipe breaks in the auxiliary building since the Kewaunee licensing basis for the shutdown cooling mode of operation includes no discussion regarding protection from the effects of random, moderate-energy pipe breaks.

Question 3:

> Is RHR operability mode specific? If the RHR system is inoperable below hot shutdown, is the system inoperable above hot shutdown? If it is determined that the RHR system is operable above hot shutdown, but inoperable below hot shutdown, what would be the required licensee action?

SBPB Response:

The basis for determining operability is mode-specific because some design-basis events are credible only in certain operating modes, and operability is based on the capability to complete the required system function during design-basis events. The NRC technical guidance regarding operability describes that requirements based on safety analyses do not necessarily have to be the same for all modes of operation. The required actions are specified by the relevant technical specification. In the case of the RHR system, operability in the cold shutdown and refueling modes of operation is based on the shutdown cooling function, and operability above the hot shutdown modes is based on the emergency core cooling and long-term, post-accident recirculation cooling functions. These functions have different safety analyses, so inoperability in one mode does not translate to inoperability in another mode. Operability is determined by evaluating functional capability in each mode.

If the RHR system is inadequately protected from failures of non-seismic piping (i.e., the system protection is degraded to the extent that the system is inoperable), the plant must be maintained in a mode where the RHR function could be performed by other systems that are adequately protected from failures of non-seismic piping. This translates to maintaining the plant in the hot shutdown mode, where the LOCA mitigation functions are not required and the RHR function can be performed by the steam generators and the auxiliary feedwater system.

In summary, the answers to the multi-part Question 3 are as follows:

Part 1: Is RHR operability mode specific?

"Yes"

Part 2: If the RHR system is inoperable below hot shutdown, is the system inoperable above hot shutdown?

"Not necessarily" since inoperability in one mode does not translate to inoperability in another mode.

Part 3: If it is determined that the RHR system is operable above hot shutdown, but inoperable below hot shutdown, what would be the required licensee action?

If the plant is in hot shutdown or above, it must remain in a mode other than cold shutdown or refueling until operability in those modes is restored. In the case of inoperability below hot shutdown, the only option is to promptly restore the system to operable status since it is unreasonable to require the licensee to increase the operational mode level. Operability may be restored by seismically qualifying the piping, providing protection for one train from the potential failure, or isolating the flow to the unqualified piping segment.

Principal Contributor: S. Jones

Date:

Reference 12

Pioneer Engineering & Services Memorandum/Telephone Log, Protection of Reactor Protection System and Engineered Safety Features from Pipe Rupture, Jet Impingement, or Pipe Whip Reactions," dated September 25, 1972

CA 705119
MS Analysis

CA ~~703430~~ 705118
w Analysis

CA 700139

Pioneer Service & Engineering Co.

MEMORANDUM/TELEPHONE LOG

[illegible]: PRAIRIE ISLAND NUCLEAR GENERATING PLANT - UNITS 1 & 2 NO: 21-6197
PROJECT: KEWAUNEE NUCLEAR POWER PLANT NO: 23-7127

[illegible]: As Indicated DATE: September 25, 1972

[illegible]: Jay S. Iyer/ E. U. Claeson S. F. NO: 300.3410
300.3610

[illegible]: As Indicated

C. Agan M. DeLong M. Armando M. T. Lin

[illegible]	C Didler	H Gryzbac	K Bollmeier	C LaRoche	B Musial	D Sahlin	S Tarabus
[illegible]	[illegible] Dodt	[illegible] Grzelakowski	K Boyer	L LaTourette	I Nelson	A Sapphire	F Tomaselli
[illegible]	H Drever	A Gottman	A Ibe	J Leipper	L Newhart	H Scholes	J Toresmo
[illegible]	[illegible] Edelstein	J Hajek	S Inouye	S Lemon	V Notson	R Schuler	E Touchette
[illegible]	E Elwood	D Hamady	J Iyer	A Lenzi	G Ong	C Schwanderlik	J Towle
[illegible] Bradley	I Emms	D Hanson	K Johannsen	D Leppke	D Peaks	R Small	J Tully
[illegible]	J Fegan	A Hasnet	E Johnson	L Lindner	F Peterson	D Smiley	A [illegible]
[illegible] Briggs	M Fencke	C Hassell	R Kallenbach	E Loukota	C Pirofalo	G Soukup	G Vellender
[illegible]	H Fossum	E Haupt	E Kandra	W Lowry	J Poer	D Sparks	R Vojta
[illegible]	F Furrer	R Hawes	T Kitz	K Malsy	E Purcell	H Spitzer	R Weatherhead
[illegible]	R Furrer	P Hawley	T Klotz	G Matt	E Prahin	A Sterlin	R Weaver
[illegible]	E Gautchier	J Hayes	A Koehler	J May	A Rachofsky	K Stratton	A Weiss
B Cooper	W Gelszinnus	H Herman	R Larson	T McGovern	W Raley	R Straus	A Wong
[illegible]	L Goslovich	F Hickey	J Laspisa	R Mitchell	W Rieger	E Streich	E Zimmerman
H Crumpacker	J Goy	H Hirschhorn	M Laspisa	P Morano	L Ryan	D Swanson	
[illegible] Deutsch	E Griffith	H Hollingshaus					

SUBJECT: Protection of Reactor Protection System and Engineered Safety Features from Pipe Ruptures, Jet Impingement or Pipe Whip Reactions

The attached study (separate attachment to this memorandum) essentially outlines the criteria to be used for Protection of Reactor Protection System and Engineered Safety Features from Pipe Ruptures, Jet Impingement or Pipe Whip Reactions and applies to both Kewaunee and Prairie Island Projects.

At a meeting on Friday, September 22, 1972 at 1:00pm in Conference Room A, when the above addressees were present, the following conclusions given below, were reached regarding the criteria and the immediate work effort required (as indicated below) on the Prairie Island Project.

A. Criteria: NSP has accepted the DRL position on this subject and will essentially abide by the requirement spelled out in the DRL Safety Evaluation. A preliminary copy of a telephone conversation on this was given to all the above personnel. If not before at the ACRS sub-committee meeting on October 24,1972 NSP will state that they will make necessary modifications to the plant (as far as practicable) to meet this requirement. The AEC-DRL position is restated below:

Do NOT Destroy

PROJECT FILE 23 - 7127 Date 9-25-72 File #

"We conclude that the applicant's criterion should be modified to state that no single event will cause failure of redundant reactor protection system circuits or ESF components, in a manner such that a single failure after the event prevents the protective function of associated ESF."

We interpret the criteria as follows:

"No single event (e.g. pipe rupture-steam line) will cause failure of Reactor Protection System or Engineered Safety Features required (needed) to mitigate the consequences of this event, in a manner such that a single failure after the event prevents required protection".

To elaborate a little further this criterion requires that all systems required to mitigate the consequence of an accident be designed such that the accident does not take out of service either of a redundant two train systems. It is understood that during maintenance periods on protection systems or ESF this criterion is violated. However the Technical Specifications spell out the time period for equipment out on maintenance prior to plant shut-down.

The key word in this criterion is the list of equipment needed or required. As an example of a Main Steam Line Break, depending on the location of the break the needed equipments are different. It is therefore essential to identify the needed equipment accurately and then determine if they are affected by the accident and if so then protect the same.

B. <u>Pioneer Effort</u>: Pioneer has to come up with some answers by Friday, September 29, 1972. It is with this in mind that the meeting was called. In a smaller group meeting after this meeting, the following tasks were outlined:

(i) Mechanical (W. P. Brennan and C. Agan with appropriate D&D help) will investigate the feasibility of:

(a) A Guard-Pipe or Pipe-Chase around the Main Steam lines.

(b) Rerouting of the Main Steam lines outside the containment such that the problem is eliminated by physical separation or distance. This latter solution may also eliminate partially if not completely the necessity of a Pipe-Chase. The Structural department may be required to help in this effort as needed.

(ii) Electrical (L.C. LaTourette, B. Musial, et al) will identify the extent of the problem and the feasibility of solutions for the following three areas:

(a) Main Steam Lines inside the Containment.

(b) Feedwater Lines inside the Containment.

(c) Feedwater Lines outside the Containment.

For Electrical to proceed in the above three efforts they will be provided with three separate lists of equipment (ESF) required to mitigate the consequences of these accidents. Until this list is ready Electrical is requested to proceed with the list provided by C. Agan of 8/18/72 for the "Main Steam Line Break". This list includes all of the equipment required for a break inside the containment and will therefore be the same list for item (b) above.

(iii) Nuclear Analysis (M.Lin) will proceed to calculate compartment pressures (outside the containment) for a feedwater line break.

(iv) Structural will proceed to determine the strength of walls, doors, etc. for withstanding a Main Steam Line rupture outside the containment.

(v) In addition to the above, the rupture anywhere in the Reactor Coolant Pressure Boundary will be investigated and the expected people to participate are Electrical, Mechanical and I & C. I will try to get this last task underway myself.

(vi) The Analytical Section (R.J. Hollmeier's Group) is requested to help Electrical immediately regarding information on Jet Forces, distances, etc. for items (ii) (a), (b), and (c) in a form that they can easily utilize.

Please prepare a write up for items (i) (a), and (i) (b), by W. P. Brennan/C. Agan, (ii) a, b and c by L. C. LaTourette/B. Musial/ A. Sapphire. The write up should include, but not limited to, a discussion of the investigations, results of solutions. Cost estimate, Schedule impact, Engineering mandays for detail and design, etc. If you have any difficulties, please let me know immediately.

Please note that the study mentioned in the first paragraph will be forthcoming separately. It is mandatory that we have reasonably firm solutions by Friday September 29, 1972 so that this can be reviewed with NSP during the week of October 2, 1972. NSP has presently planned a meeting with AEC-DRL (R.C. DeYoung etc.) on October 11, 1972 and therefore it makes it all the more important to meet our dates.

In addition we will have to investigate the entire plant to verify that the criteria can be met and if not, necessary modifications will have to be made to the plant design. This effort of course, will be undertaken right after the completion of the above listed tasks.

Reference 16

NUREG-1174, "Evaluation of Systems Interactions in Nuclear Power Plants, Technical Findings Related to Unresolved Safely Issue A-17," dated May 1, 1989 (Only Appendix Included)

APPENDIX
INTERNAL FLOODING AND WATER INTRUSION INSIGHTS

Operating events have demonstrated the susceptibility of individual plant components to water intrusion and flooding from internal plant sources. Flooding, as discussed here, includes flooding of equipment by large volumes of water (i.e., equipment submergence) and other forms of water intrusion, including water spraying, dripping, or splashing on sensitive equipment. Examples of these types of events can be found in an operating experience review (References 1 and 2) conducted by the NRC and in individual NRC information notices (References 3–9). A key point apparent from these events is that the quantity of the water involved is not necessarily a measure of the problems that the water can create; the *location* of the water is much more significant. For example, a small leak that drips down through electrical equipment can have a more severe impact on the plant than an 8-foot flood in a pump compartment. Also, Generic Issue 77, "Flooding of Safety Equipment Compartments by Back-Flow Through Floor Drains," has received a high priority ranking (Reference 10) because of the possibility that plant designs have overlooked backflow through floor drains as a flooding pathway.

All plants should have conducted some flooding-type studies as part of demonstrating conformance to various requirements. These requirements were typically focused on large volumes of water and the potential for submerging equipment.

(1) The general design criteria (10 CFR Part 50, Appendix A) address the area of flooding. Specifically:

- GDC 3, "Fire protection," states: "Fire fighting systems shall be designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of these structures, systems and components designated as important to safety."

- GDC 4, "Environmental and dynamic effects missile design bases," states: "Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with...normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These structures, systems and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit. However, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping."

(2) As part of environmental qualification requirements of 10 CFR 50.49, submergence was evaluated for certain equipment for water associated with design-basis events.

(3) Generic letters issued to licensed facilities in 1972 required additional review based on an event at the Quad Cities plant.

(4) For more recently licensed plants, the Standard Review Plan (Reference 11) cites the generic letters of 1972, and therefore, flooding-type analysis should have been performed as part of the licensing process.

In addition, all plants should have developed programs for the review of operating experience per the requirements of Item I.C.5 of NUREG–0737 (Reference 12). These reviews should include consideration of NRC information notices and other industry documents such as those issued by the Institute of Nuclear Power Operations (INPO). Both of these have included events involving flooding and water intrusion.

The staff has concluded that existing requirements lack specific guidance regarding water intrusion events that may involve small amounts of water and subtle paths of communication of water or moisture to sensitive equipment.

The staff also recognizes that it may not be possible to identify all subtle pathways and sources. However, the staff believes that risk could be reduced significantly by conducting a focused review that includes:

(1) reviewing actual industry operating experience involving water intrusion for applicability to the licensee's plant

(2) considering action such as sealing conduit or providing shields for sensitive equipment, and

(3) examining safe-shutdown equipment specifically focusing on the potential for water intrusion problems. Safe-shutdown equipment for a flooding or water intrusion event would typically include the equipment needed to perform the following functions:

- Bring the plant to hot shutdown and establish heat removal.

- Maintain support systems necessary to establish and maintain hot shutdown.
- Maintain control room functions and instrumentation and controls necessary to monitor hot shutdown.
- Provide alternating current and/or direct current emergency power as needed on a plant-specific basis to meet the above three functions.

[*Note:* In addition to the above equipment, a review should include electrical equipment that could cause inadvertent actuation of components which in turn could hinder the ability to perform these functions (e.g., logic cabinets that actuate the automatic depressurization system).]

On the basis of a large amount of industry experience, the staff has determined that a flooding (including water intrusion) analysis should address the aspects listed below. Water intrusion includes all forms of water or moisture release from water sources internal to plant structures (e.g., leaks or ruptures of water or steam sources or from fire-suppression system actuation). Regardless of the means of release, the failure mechanism is intrusion of water or moisture to sensitive equipment (e.g., electrical cabinets).

(*Note:* If an analyses has been performed to demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping (i.e., per revised GDC 4), then fluid discharge associated with that rupture may be excluded from further consideration.)

Water Intrusion Considerations

Sources

The water can and has been released by failure (e.g., leaks, ruptures), by system actuation (e.g., fire-suppression system), or by special plant situations during maintenance or testing. Actual operating experience has demonstrated problems that emanate from:

- domestic water systems (toilets, sinks, eye-wash stations, etc.)
- fire-suppression equipment
- moderate-energy piping systems such as circulating water
- maintenance actions (e.g., draining, venting)
- low-pressure steam and condensate leakage

Pathways

Operating experience has demonstrated that separate rooms do not necessarily provide protection because of

- drain systems that may be plugged or allow backflow
- heating and ventilation ducts and penetrations between rooms
- unsealed doors
- unsealed or inadequately sealed electrical conduit and penetrations (either by design or from inadequate maintenance)
- unusual maintenance situations (temporary drain lines, water barriers)

Operating Experience

Collective industry experience has been described in:

- NRC Information Notice 83–41, "Actuation of Fire Suppression System Causing Inoperability of Safety-Related Equipment," June 22, 1983
- NRC Information Notice 83–44, "Potential Damage to Redundant Safety Equipment As a Result of Backflow Through the Equipment and Floor Drain Systems," July 1, 1983
- NRC Information Notice 85–85, "Systems Interaction Event Resulting in Reactor System Safety Relief Valve Opening Following a Fire-Protection Deluge System Malfunction," October 31, 1985
- NRC Information Notice 86–106, Supplement 2, "Feedwater Line Break," March 18, 1987
- NRC Information Notice 87–14, "Actuation of Fire Suppression System Causing Inoperability of Safety-Related Ventilation Equipment," March 23, 1987
- NRC Information Notice 87–49, "Deficiencies in Outside Containment Flooding Protection," October 9, 1987
- NRC Information Notice 88–60, "Inadequate Design and Installation of Watertight Penetration Seals," August 11, 1988

REFERENCES

1. U.S. Nuclear Regulatory Commission, NUREG/CR–3922, "Survey and Evaluation of System Interaction Events and Sources," Vols. 1 and 2, January 1985.

2. ———, AEOD/C402, "Operating Experience Related to Moisture Intrusion in Electrical Equipment at Commercial Power Reactors," June 1984.

3. ———, Information Notice 83-41, "Actuation of Fire Suppression System Causing Inoperability of Safety-Related Equipment," June 22, 1983.

4. ———, Information Notice 83-44, "Potential Damage to Redundant Safety Equipment As a Result of Backflow Through the Equipment and Floor Drain Systems," July 1, 1983.

5. ———, Information Notice 85-85, "Systems Interaction Event Resulting in Reactor System Safety Relief Valve Opening Following a Fire-Protection Deluge System Malfunction," October 31, 1985.

6. ———, Information Notice 86-106, Supplement 2, "Feedwater Line Break," March 18, 1987.

7. ———, Information Notice 87-14, "Actuation of Fire Suppression System Causing Inoperability of Safety-Related Ventilation Equipment," March 23, 1987.

8. ———, Information Notice 87-49, "Deficiencies in Outside Containment Flooding Protection," October 9, 1987.

9. ———, Information Notice 88-60, "Inadequate Design and Installation of Watertight Penetration Seals," August 11, 1988.

10. ———, NUREG-0933, "A Prioritization of Generic Safety Issues," December 1983.

11. ———, NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," LWR edition, July 1981.

12. ———, NUREG-0737, "Clarification of TMI-2 Requirements," September 1980.

Attachment 19

Letter from S. A. Varga (NRC) to D. C. Hintz (WPSC), dated September 6, 1985.

K-85-195

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

September 6, 1985

Docket No. 50-305

Received 9/18/85

Mr. D. C. Hintz
Manager - Nuclear Power
Wisconsin Public Service Corporation
Post Office Box 19002
Green Bay, Wisconsin 54307-9002

Dear Mr. Hintz:

By letter dated November 4, 1983, the NRC issued a Draft Safety Evaluation addressing the low temperature overpressure protection (LTOP) system at the Kewaunee Plant. We stated concerns about the LTOP alarm system and the test program for the system. You responded to our concerns in letters dated December 16, 1983, May 17 and October 30, 1984.

As indicated in the enclosed Safety Evaluation we have found your analysis and modifications of the Kewaunee LTOP system to be acceptable. This letter completes action on our TAC No. 06886. The issuance of amended Technical Specifications will be the subject of a separate review.

Sincerely,

Steven A. Varga

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Enclosure:
As stated

cc w/enclosure:
See next page

Mr. D. C. Hintz
Wisconsin Public Service Corporation

Kewaunee Nuclear Power Plant

cc:
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Kewaunee, Wisconsin 54216

Mr. Donald L. Quistroff, Chairman
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Chief Engineer
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SAFETY EVALUATION REPORT

KEWAUNEE NUCLEAR POWER PLANT

LOW TEMPERATURE OVERPRESSURE PROTECTION SYSTEM

1.0 INTRODUCTION

By letter dated August 11, 1976, the NRC requested Wisconsin Public Service Company (the licensee for the Kewaunee plant) to analyze the susceptibility of the reactor coolant system (RCS) to low-temperature overpressurization events, to propose procedural improvements to reduce the likelihood of such events, and to propose design modifications to mitigate the consequences of such events. By letter dated October 15, 1976, and supplementary letters, the licensee responded to this request.

Initially, the licensee proposed to install a new instrumentation and control system that would actuate the two power-operated-relief-valves (PORV's) on the pressurizer. However, if one PORV is taken as the single failure, the relief capacity of the remaining PORV is insufficient to accommodate the worst-case postulated overpressurization transient. Therefore, the licensee changed the approach to using spring-loaded safety valves on the residual heat removal (RHR) system. While the safety valves are passive in that no instrumentation and controls are directly involved, the RHR isolation valves (that isolate the RHR from the RCS) must be open to provide access to the safety valves. The instrumentation and controls associated with the RHR isolation valves, therefore, become involved with the low temperature overpressure protection system (LTOPS). The LTOPS also includes certain alarm features.

By letter dated October 24, 1978 and supplementary letters, the licensee proposed a license amendment (No. 35) to provide Technical Specifications for the LTOPS.

This evaluation report addresses the instrumentation and controls aspects of the LTOPS design and Technical Specifications.

2.0 SYSTEM DESCRIPTION

There are two 8-inch drop lines which connect the two RCS hot legs to the common RHR suction line. The RHR suction line has a 2-inch safety valve now set to open at 480 psig. The licensee proposed, and has since installed, a new 4-inch spring-loaded safety valve on the RHR suction line with a setpoint of 500 psig. This configuration is shown in Figure 1. The safety valves are passive in that there are no instrumentation/controls or external power sources directly involved. When the RCS is at low temperatures (i.e., below 342°F) and the RHR is connected, the spectrum of postulated pressure transients would be mitigated, first by the 2-inch safety valve and as necessary by the 4-inch safety valve, such that the temperature-pressure limits of Appendix G to 10 CFR 50 would not be exceeded.

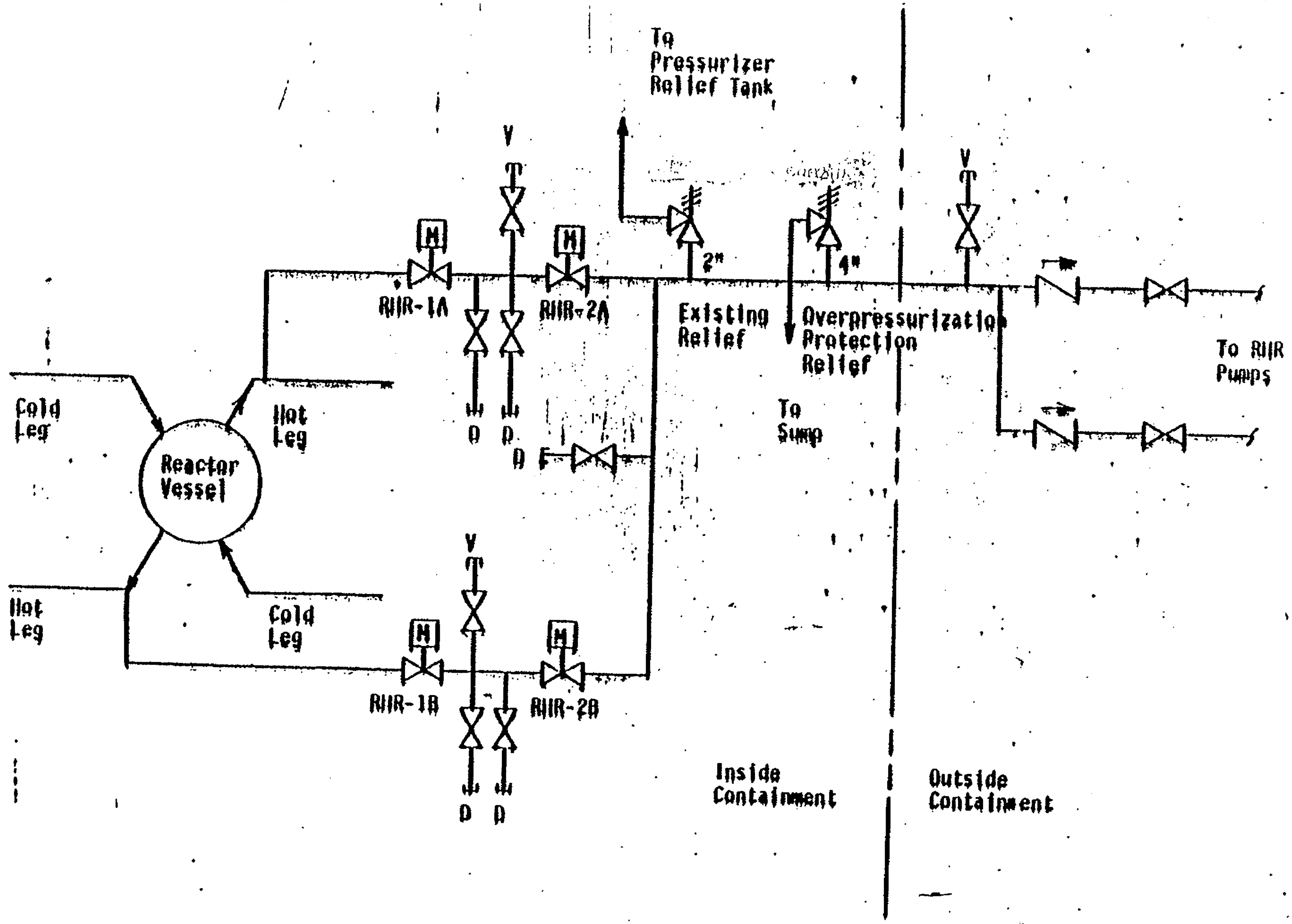

Figure 1. Flow diagram of residual heat removal system.

Each drop line has two motor operated RHR isolation valves (MOV's) in series, which must be opened and remain open to provide access to the LTOPS safety valves. There are instrumentation and controls (referred to as the "RHR interlocks") associated with these 4 MOV's. Below a pressure of 450 psig, a permissive signal is generated that permits the manual opening of the MOV's.

Additionally, the licensee proposed (and subsequenlty impemented) a new alarm feature called "RHR improper (valve) lineup". This annunciator alarm is presently actuated in either of two conditions: First, if any RHR isolation valve is not-fully-open when RCS pressure is below 450 psig. Second, if any RHR isolation valve is not-fully-closed when RCS pressure is above 700 psig.

3.0 EVALUATION

The basic safety objective is that no single equipment failure or operator error will result in exceeding the temperature-pressure limits of Appendix G to 10 CFR 50. At the same time, consideration must be given to assure that other aspects of nuclear safety are not compromised; for example, a spurious opening of a PORV.

To implement these basic objectives, review considerations include: (1) reliance on operator actions, (2) single failure criterion, (3) testability, (4) seismic and IEEE-279 design criteria, (5) alarm features, (6) temperature/pressure recordings, and (7) Technical Specifications. Each of these areas is discussed below.

3.1 Operator Actions

The design of the LTOPS should be such that reliance upon operator actions within the first 10 minutes is not necessary. At this facility, operator actions are relied upon only to the extent that the normal operating procedures (improved to reduce the likelihood of the occurrence of an overpressurization event) are followed. To that end, certain alarm features are provided to help avoid operator errors.

If an overpressurization event occurs and the RHR is connected to the RCS, the passive LTOPS safety valves are sufficient to prevent exceeding the Appendix G limits without any operator action. Actions that an operator could take, such as securing a source of pressurization, are desirable but not essential.

Alarm features are discussed below.

3.2 Single Failure Criterion

The design of the LTOPS should be such that no single failure causes loss of RCS overpressure protection at lower operating temperatures. Failures caused by loss of instrument air or by loss of offsite electric power must be considered. The licensee elected to change to the RHR safety valve approach based, in part, on the simplicity of the system. Being spring-loaded safety valves, the LTOPS does not depend directly either upon instrumentation channels (consisting of sensors, transmitters, bistables, relays, solenoids, valve operators, etc., that could fail) or upon external power sources. Therefore, no directly related instrumentation or controls failures or loss of power affect the LTOPS performance.

The instrumentation/controls and power aspects of the RHR interlocks of the isolation valves must be considered. The licensee recognized that, in the previous RHR interlock design, a single failure of either pressure channel could have caused the LTOPS safety valves to become isolated from the RCS, and has made an appropriate design change. Further, the licensee has performed a single failure analysis. The analysis encompassed such items as postulated failures of either pressure channel to either the high pressure or low pressure state and operator errors.

On loss of electric power, the pressure transmitters fail low and therefore would not have a effect on LTOPS. In our review, we also considered the possibility of a single failure exposing the low-pressure piping of the RHR to the higher pressure of the RCS. With the change in the logic of the RHR interlocks to a per-drop-line arrangement, a failed-low transmitter completes the logic for a pathway connecting the RHR and RCS. The competing single failure criterion objectives for both LTOPS protection and RHR protection have spawned the suggestion that four pressure transmitters might be needed. If a failed-low transmitter actually opened the RHR isolation valves, four transmitters should be provided. However, action of the failed-low transmitter is only to generate a permissive for valve opening. In addition to the equipment failure, operator actions contrary to both procedures and training are necessary to expose the RHR piping. Therefore we find that the failure of a single pressure transmitter is acceptable and that additional transmitters are not necessary. Since only one RCS drop line is affected by a single transmitter and an alarm is generated, this failure result is acceptable. There were no single failure events identified that could cause the LTOPS safety valves to be isolated from the RCS. Accordingly, based upon our review of the licensee's failure analysis, we conclude that no single failure in the instrumentation or controls will prevent the performance of the LTOPS.

We also reviewed the LTOPS to ascertain if a single failure could cause spurious operation of the LTOPS during power operations. The worst failure would involve the opening of the RHR isolation valves. Since pressure transmitters' failing low (on loss of power) results in only a permissive for the opening of the RHR isolation valves, and not automatic MOV opening, we conclude that no single failure can result in spurious operation of the LTOPS safety valves during power operations.

During this review, it was noted that motive power to the RHR isolation valves is provided by two non-Class 1E sources and aligned on an inboard-outboard MOV basis. If offsite power is lost, motive power to these RHR valves would be lost. However, being MOV's, these valves fail in the "as-is" position on loss of power. Therefore, if RHR operation is in progress (with the RHR valves open), loss of offsite power has no effect on LTOPS performance. If offsite power is lost before RHR operation is attained, such a changeover to RHR would be inhibited at least temporarily. The Kewaunee plant was licensed on the basis that reaching hot shutdown status is sufficient. The use of RHR is not required for hot shutdown and at hot shutdown temperatures the existing PORV's and safety valve provides adequate RCS overpressure protection. Further, long-term core cooling following an accident is provided by the recirculation mode of the emergency core cooling system.

Pursuant to Appendix R to 10 CFR 50, the licensee has made significant plant modifications that will assure that cold shutdown can be reached within 72 hours. The licensee has changed the motive power scheme for the RHR isolation valves as part of the Appendix R program. Motive power is obtained from separate Class 1E sources and assigned to the MOV's on a per-drop-line basis.

We conclude that the licensee's actions relative to providing motive power to the RHR isolation valves are appropriate and acceptable.

3.3 Testability

The LTOPS design should be testable on a periodic basis commensurate with the required operability of the LTOPS. The licensee will be testing/calibrating the LTOPS safety valves periodically as part of the Inservice Inspection program.

The RHR interlock system is presently tested by means of two surveillance procedures. SP 34-145C Rev. C provides that the pressure transmitters PT 419 and PT 420 are calibrated during each refueling outage. This procedure assures that for various input pressures (in the range of 0-3000 psig) the output current is the proper valve (in the range of 10-50 mA). The acceptance band is $\pm$ 0.2 mA, which corresponds to $\pm$ 0.5% of full scale or $\pm$ 15 psig. SP 34-145D Rev. D provides for a calibration check of the 700 psig bistable setpoint, functional test of the auto-closure feature when the 700 psig setpoint is reached, functional test of the open permissive when the 450 psig setpoint is reached, and functional test of the RHR improper valve lineup alarm. This SP is also performed on a refueling outage basis.

At the start of each plant cooldown for refueling, the most recent test/calibration of the RHR interlocks is very old --- having been performed only at the previous refueling outage. In this time a malfunction or failure could have occurred. We discussed with the licensee that feasibility of testing during power operation just prior to plant cooldown. The licensee expressed concern regarding cycling the RHR valves on a one-at-a-time basis due to the possibility of pressuizing the water volume between the valves and subsequently releasing this pressure onto the the RHR. It appears, therefore, that the "cost" of testing the RHR isolation valves may outweigh the benefits.

We conclude the licensee has acceptable testability procedures in place. We are currently reviewing a license amendment which proposes Technical Specifications in the area of fracture toughness related to relief valve set-points, an issue separate from this SER.

3.4 Seismic and IEEE 279 Criteria

This review criterion was established generically for PWR's and stated that, ideally, the LTOPS system should meet seismic Category 1 and the IEEE 279 criteria for protection systems equipment. This LTOPS criterion was established based upon meetings with PWR vendors and owners groups. It appeared at that time that most licensees would adopt the approach of installing a new instrumentation system to control the existing PORVs at reduced pressure setpoints. A new system designed today should meet today's standards, which might include seismic Category 1 criteria and the IEEE 279 criteria.

In this case, however, a new instrumentation system is not being installed and, in fact, the LTOPS is passive and does not include an instrumentation channel. The licensee has however modified the existing control-grade instrumentation system for the RHR interlocks. When an existing control system is modified, it is not always clear to what extent seismic and IEEE 279 criteria should be invoked, since it would be a backfit. The primary criterion in IEEE 279 is the single failure criterion. For this pariticular case, we have determined that if the single failure criterion is satisfied, this constitutes a sufficient application of the IEEE 279 criteria.

The licensee has stated that the RHR interlock system is not fully qualified as seismic Category 1. The licensee has presented his argument that the RHR interlock should not be required to be backfitted to seismic Category 1 criteria. We believe that to postulate a design basis event that involves an overpressurization transient concurrent with a seismic event of the magnitude of the safe shutdown earthquake would cause more backfitting then is warranted in this case. Therefore we conclude that satisfaction of the single failure criterion is sufficient.

3.5 Alarms

In our letter of January 16, 1985 we approved removal of the auto closure feature from the valves that isolate the suction side of the Residual Heat Removal System at Kewaunee. This action, in addition to the licensee's letter of December 16, 1983, responded appropriately to our concerns of November 4, 1983 (Draft SER) regarding alarms at Kewaunee.

In general, control room alarms are provided to help avoid operator errors. In this case, an alarm would alert the operator to take action (i.e., serve as a trigger point) when low temperature overpressure protection is required.

Two problems have been identified by our November 4, 1983 review of the "RHR improper (valve) lineup" annunciation. The inputs to the alarm are RCS pressure and RHR loop isolation valve position indication. This alarm was initially intended to provide the operator solely with information to protect RHR system integrity. However, it acquired a dual purpose upon backfitting the LTOP requirement at KNPP. As such, there are times during normal heatup or cooldown evolutions when this alarm may be actuated but invalid. We suggested that RCS temperature replace pressure as the alarm input as a means to limit the range over which the alarm is valid.

The present design will allow an invalid alarm for a short period of time. However, this alarm state is an expected condition at Kewaunee for a controlled transient (heat up from cold shutdown or cool down to a cold shutdown). As such it does not constitute an ambiguous alarm. To provide additional assurance that the alarm is understood, a cautionary note dealing with this alarm has been added to the appropriate procedures.

The annunciator system, including this alarm, is being studied as part of the Kewaunee Detailed Control Room Design Review (DCRDR). This program will provide a structured review of the annunciator system from an overall systems standpoint. Human Engineering Observations identified by this review will be evaluated by the CRDR team, which will then make recommendations for resolution in accordance with the CRDR plan. We agree to this approach.

The second problem identified by NRC review is that there is no direct indication that a pressure transient is occurring. We suggested the licensee install an acoustical monitor to indicate flow through the safety valves. The licensee in their November 4, 1983 letter points out direct indication is available. The RCS pressure channels are a direct indication of RCS response to pressure transients. A pressure transient is recognized as a steady increase in system pressure which stabilizes or cycles about the relief valve setpoint, impervious to operator actions to raise pressure further. This pressure response is abnormal and would be recognized by the control operator. Additionally, the 2-inch relief valve discharges to the pressurizer relief tank (PRT), so that PRT level and pressure also serve as indication of relief valve actuation. In addition, the deletion of the auto closure feature helped resolve this concern.

We conclude that the above direct indication of LTOP System actuation is acceptable.

3.6 Pressure/Temperature Recordings

The staff criteria for LTOPS includes providing a permanent pressure/temperature record of any pressure transient. The response time of such a recorder should be compatible with pressure transients with a rate of 100 psig/second. The licensee has installed wide range temperature and pressure monitoring equipment which provides a continuous record of all plant conditions.

3.7 Technical Specifications (TS)

The licensee has submitted TS for our review. The review is separate from this SER and relates to the protection of the fracture toughness integrity of the reactor coolant system and RHR system.

4.0 SUMMARY

The licensee has switched the design approach for the low temperature overpressure protection system (LTOPS) from a new instrumentation system that would actuate the PORV's on the pressurizer to passive spring-loaded safety valves on the common RHR suction line. The previously existing interlock features for the RHR isolation valves have been modified to assure that the safety valves will be available for the LTOPS function. Control room alarms/annunciators and pressure/temperature recorders have been provided.

The LTOPS is self-sufficient in that it does not rely upon external power sources or operator action when the RHR system is connected to the RCS. A single failure analysis has been performed and concluded that no failure in the instrumentation or controls will prevent the functioning of the LTOPS. We have determined that no single failure during power operation will cause spurious operation of the LTOPS. We have determined that, since an existing control-grade instrumentation system was modified (as contrasted to the design and installation of a new system), satisfying the single failure criterion is a sufficient degree of backfit application of IEEE Standard 279.

5.0 CONCLUSION

We conclude, based on the above evaluation and our January 16, 1985, Safety Evaluation that removed the auto closure feature, that the KNPP low temperature overprotection system is acceptable.