WILLIAM L. BERG
President and CEO

DAIRYLAND POWER
COOPERATIVE

May 19, 2010

In reply, please refer to LAC-14117

DOCKET NO. 50-409

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

SUBJECT: Dairyland Power Cooperative
La Crosse Boiling Water Reactor (LACBWR)
Request for Amendment to 10 CFR 50, Possession-Only License in Support of Dry Cask Storage Project – Response to NRC Request for Additional Information

REFERENCES:
1. LACBWR Possession-Only License No. DPR-45, Docket No. 50-409, Amendment No. 69, and Appendix A, Technical Specifications, Amendment No. 70.
2. LACBWR Decommissioning Plan, Revised December 2009.
3. LACBWR Request for License Amendment dated July 28, 2009
4. NRC Request for Additional Information, dated March 11, 2010

In the Reference 3 correspondence, Dairyland Power Cooperative (DPC) submitted a license amendment request (LAR) proposing changes to the LACBWR License Appendix A, Technical Specifications (TS). These changes were requested in support of the LACBWR Dry Cask Storage Project that will establish an Independent Spent Fuel Storage Installation under general license provisions of 10 CFR 72, Subpart K. These TS changes were requested to accommodate efficient dry cask storage system (DCSS) loading operations and reduce overall occupational dose to personnel during these operations. In Reference 4 the NRC provided a Request for Additional Information (RAI) to DPC containing a number of questions and requests for information pertaining to the LAR. This submittal provides the responses to those RAIs.

In preparing the responses to the RAIs, DPC re-evaluated the LAR and the estimated dose reduction benefit of the requested changes. As a result of this re-evaluation, DPC has decided to reduce the scope of the LAR in order to increase the operating safety margin during cask loading, while still gaining operational efficiency and personnel dose reduction benefits. The LAR now proposes only a reduction of the minimum water coverage over spent fuel from 16 feet to 11 feet, 6½ inches, and a small number of editorial clarifications to clarify heavy load controls and reflect inclusion of the cask pool in the scope of TS as part of an "extended" Fuel Element Storage Well (FESW). The primary result of this decision is that proposed new LCO 4.1.2.c is being removed from the LAR.

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The reduction of the LAR scope described above will now require the fuel transfer canal gate to be removed and reinstalled as necessary for dry run training and cask loading operations until the upper tier fuel storage racks in the FESW no longer contain any spent fuel assemblies. Once all fuel assemblies are out of the upper tier racks, installation of the fuel transfer canal gate is not required to comply with the proposed 11 feet, 6½ inches of water coverage over spent fuel assemblies. Please refer to the response to RAI 2 for a more detailed discussion of the operational sequence, including water levels and fuel transfer canal gate requirements.

The responses to the RAIs are written with consideration of the reduction in LAR scope. A complete set of replacement proposed revised changes to the Technical Specifications are also provided with this submittal to supersede those submitted with Reference 3.

Based on our evaluation, DPC concludes that the responses to the RAIs and the reduction in LAR scope have no impact on the no significant hazards consideration previously provided with the Reference 3 submittal.

As previously discussed with the NRC, dry cask storage loading operations are now expected to begin in July 2011, which is approximately one year later than previously anticipated when Reference 3 was submitted. Therefore, we have modified our request date for approval of this license amendment to December, 2010. This date provides additional review time for the NRC staff and still meets our need to have the amendment approved prior to pre-loading campaign training activities.

If you have any questions regarding this license amendment request, please contact LACBWR Plant Manager Mike Brasel at (608) 689-4220.

Sincerely,

DAIRYLAND POWER COOPERATIVE

William L Berg

William L. Berg
President and CEO

WLB:JBM: two

Attachments:

1. Responses to RAIs with attachments.

2. Technical Specification Pages as Changed.

cc: Mr. Mark Satorius, Regional Administrator, Nuclear Regulatory Commission Region III
Ms. Kristina Banovac, NRC Project Manager
Paul Schmidt, Manager, Radiation Protection Section, State of Wisconsin

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STATE OF WISCONSIN)
)
COUNTY OF LA CROSSE)

Personally came before me this 25th day of May, 2010, the above named, William L. Berg, to me known to be the person who executed the foregoing instrument and acknowledged the same.

Laurie A. Engen
Notary Public, La Crosse County Wisconsin

My commission expires August 8, 2010

LAURIE A. ENGEN
Notary Public
State of Wisconsin

Dairyland Power Cooperative (DPC)
La Crosse Boiling Water Reactor (LACBWR)

Response to Request for Additional Information
Regarding Request for Amendment to 10 CFR 50, Possession-Only License in
Support of Dry Cask Storage (DCS) Project

Attachment 1

Questions and Responses

May 19, 2010

RAI 1

The license amendment request (LAR) describes the licensee's plans for loading spent fuel into casks for on-site dry cask storage. The licensee is planning to use the general license provisions in 10 CFR Part 72, Subpart K, for storage of spent fuel at its site. A cask pool will be constructed in the area that previously housed the reactor pressure vessel, and the existing fuel transfer canal will be used to move the fuel from the FESW to the cask pool where it will be placed in an NRC-certified (under Part 72) spent fuel storage canister. During fuel loading operations, the fuel transfer canal and cask pool will be hydraulically contiguous with the FESW, and can be considered as an extension of the FESW.

The LAR does not specifically address the design of the cask pool. NRC staff understands that the licensee is currently evaluating whether the design and construction of the cask pool requires prior NRC approval, per 10 CFR 50.59.

The current TS (for the 10 CFR Part 50 license, DPR-45) contain requirements for the design and conditions of the FESW and for fuel storage and handling. The LAR does not address how these TS requirements apply to the "extended FESW," including the cask pool.

The LAR discusses the "jurisdiction of 10 CFR [Part] 72 over [dry cask storage system] DCSS loading operations," in relation to proposed changes to the definition of "fuel handling." It appears that the licensee is attempting to make a distinction between handling of individual fuel assemblies and handling of a NRC-certified canister/cask that contains fuel assemblies, in this proposed change. However, the general license provisions in Part 72, Subpart K, for the storage of spent fuel at power reactor sites, do not address safety aspects of fuel handling or spent fuel pool conditions during cask loading operations. Rather, the conditions of the FESW and fuel handling and storage activities (which would include activities within the "extended FESW") are governed by Part 50. The requirements in the Part 50 license and TS continue to generally apply to the conditions of the FESW and to fuel storage and handling, during cask loading operations.

Please explain how the Part 50 TS, related to design and conditions of the FESW and fuel storage and handling, will apply to the "extended FESW," or propose appropriate TS controls to address the "extended FESW," taking into consideration the conditions of the Certificate of Compliance for the cask that will be used at LACBWR.

Response to RAI 1

The NAC International MPC System (NAC-MPC) FSAR (currently nearing the end of the NRC review to approve an amendment to Certificate of Compliance (CoC) 1025 for use at LACBWR) refers to the "spent fuel pool" in generic terms. This generic term customarily applies to the cask user's spent fuel pool, which may include a cask loading area within the spent fuel pool, or an adjacent cask loading pit connected to the main spent fuel pool by an opening in the common wall to permit spent fuel assembly movement between the two. At LACBWR the spent fuel pool is known as the Fuel Element Storage Well (FESW). The LACBWR FESW contains 333 spent fuel assemblies in a two-tiered, U-shaped rack configuration, which surrounds a cask loading area on three sides. Based on this configuration, only the FESW was discussed in the DPC License Amendment Request (LAR).

The LACBWR FESW cask loading area is sized for a small spent fuel shipping cask and will not accommodate the large-capacity spent fuel storage and shipping casks typical of today's designs, including the NAC-MPC System. The cask pool concept was conceived so that the NAC-MPC system could be used to efficiently load up to 68 spent fuel assemblies in a single storage canister. This storage canister capacity requires a total of only five storage canisters and associated vertical concrete casks to completely empty the FESW of spent fuel and transfer all of the spent fuel into dry storage at the onsite Independent Spent Fuel Storage Installation (ISFSI).

In responding to this RAI, DPC has come to agree that "spent fuel pool" in the context of the NAC-MPC FSAR should include the FESW, the fuel transfer canal, and the cask pool at LACBWR. As the RAI states, the design and installation of the cask pool will be reviewed in accordance with 10 CFR 50.59 and, if required as a result of the 50.59 review, submitted to NRC for review and approval. At this time we do not expect NRC review and approval of the cask pool design to be required.

Because DPC has re-evaluated the definition of spent fuel pool to now include the cask pool, we agree that the 10 CFR 50 Technical Specifications (TS) governing fuel handling in the FESW should be extended to include the cask pool and should apply to the activities taking place there. We have reviewed the entirety of the LACBWR TS and are proposing appropriate changes as part of this RAI response to reflect inclusion of the cask pool in the scope of the Part 50 TS.

We note that 10 CFR 50.68(c) recognizes the exclusive jurisdiction of the regulations in 10 CFR 72 for the criticality accident requirements for fuel located in an NRC-approved spent fuel storage cask while located in the spent fuel pool. This applies to the fuel in the NAC-MPC storage canister in the LACBWR cask pool. The NAC-MPC 10 CFR 72 CoC and associated TS provide appropriate controls for ensuring the criticality safety of the spent fuel assemblies while in the storage canister in the cask pool. Therefore, TS pertaining to criticality control of the spent fuel assemblies in the storage canister while situated in the cask pool are not proposed to be added to the LACBWR license. However, there are other matters that do fall under the jurisdiction of 10 CFR 50 that warrant being controlled by the LACBWR Part 50 TS such as minimum water coverage over spent fuel in the storage canister in the cask pool without the closure lid in place. To that end, we request modifications to the previously proposed changes to TS Section 1, TS 4.1, LCO 4.1.2; and for completeness, include a new change to TS Section 2.2.

These new and revised changes to the LACBWR Part 50 TS are summarized and justified individually below. The changes described below include all changes against the original, currently approved TS appended to the Part 50 license, including those that are no different than the original LAR submittal, to avoid confusion. Proposed final versions of these modified TS are also included in Attachment 2 to this submittal. These changes supersede the previously proposed changes to the TS in their entirety. The previously proposed changes to TS Section 1 that are not affected by this RAI response are still needed, and are described below and also included in Attachment 2 for clarity. Generally speaking, the term "FESW" will be referred to as "FESW and cask pool" in the areas of the TS affected by this RAI.

Proposed LACBWR Technical Specification Changes

1. TS Section 1: Definitions

The definition of FUEL HANDLING currently states:

FUEL HANDLING shall be the movement of any irradiated fuel within the Containment Building. Suspension of FUEL HANDLING shall not preclude completion of movement of the fuel to a safe, conservative position.

Proposed Change:

DPC proposes to modify this definition to read as follows (new text in bold type):

FUEL HANDLING shall be the movement of ***individual spent*** *fuel* ***assemblies*** *within the* ***Reactor*** *Building. Suspension of FUEL HANDLING shall not preclude completion of movement of* ***a spent*** *fuel* ***assembly*** *to a safe, conservative position.* ***FUEL HANDLING, for the purposes of these Technical Specifications, does not include the movement of an NRC-certified spent fuel storage canister, transfer cask, or storage cask containing spent fuel in accordance with the dry cask storage system 10 CFR 72 Certificate of Compliance.***

Technical and Regulatory Basis:

Please see Section II.1 of Attachment 1 to the original LAR submittal for a discussion of the technical and regulatory basis. In addition to the original submittal, for consistency with other sections of the TS the term "fuel" or "irradiated fuel" is replaced by "spent fuel" at three instances in this definition.

2. TS Section 2: Design Features

TS 2.2, "FUEL STORAGE"

The title of TS Section 2.2 is currently "FUEL STORAGE"

Proposed Change:

DPC proposes to revise the title of TS Section 2.2 be "FUEL STORAGE **WHILE IN THE FUEL ELEMENT STORAGE WELL"**

Technical and Regulatory Basis:

This change clarifies that the fuel storage requirements in this TS section apply only to the spent fuel assemblies in the FESW storage racks. Fuel located inside an NRC-approved spent fuel storage canister in the cask pool awaiting transfer to the ISFSI is not considered "in storage" in the context of this TS section. A corresponding change is made to the Table of Contents.

<u>TS 2.2.3, "DRAINAGE"</u>

TS 2.2.3 currently states:

> <u>*DRAINAGE*</u>
>
> *2.2.3 The Fuel Element Storage Well is designed and shall be maintained to prevent an inadvertent draining of the well below elevation of 679 ft MSL.*

DPC proposes to modify this TS to read as follows (deleted text in strikethrough and new text in bold type):

> <u>*DRAINAGE*</u>
>
> *2.2.3 The Fuel Element Storage Well is designed and shall be maintained to prevent an inadvertent draining of the well below elevation of 679 ~~ft~~* ***feet*** *MSL* ***while spent fuel assemblies are in the Fuel Element Storage Well.***

Technical and Regulatory Basis:

The proposed change to this TS is the same as the original LAR except for an editorial change to replace the term "irradiated fuel" with "spent fuel" for consistency with other sections of the TS. The intent of this TS requirement is to ensure adequate water coverage over spent fuel assemblies is maintained for shielding purposes. With no spent fuel assemblies in the FESW, there is no need for the water shielding and an inadvertent draining event would not result in loss of shielding for the spent fuel assemblies. This change allows for draining of the FESW below elevation 679 feet following removal of all of the spent fuel assemblies.

3. <u>TS Section 4/5: Performance Requirements</u>

TS 4.1.1, "GENERAL FUEL STORAGE AND HANDLING REQUIREMENTS"

a) TS 4.1.1.1 currently states:

> *4.1.1.1 Irradiated fuel assemblies shall be stored underwater in spent fuel storage racks that are positioned on the bottom of the Fuel Element Storage Well or in approved on-site dry spent fuel storage containers, or in an approved shipping cask.*

Proposed Change:

DPC proposes to modify this TS to read as follows (deleted text in strikethrough and new text in bold type):

> *4.1.1.1 ~~Irradiated~~ **Spent** fuel assemblies shall be stored underwater in spent fuel storage racks that are positioned on the bottom of the Fuel Element Storage Well or in **an** approved ~~on-site~~ dry spent fuel storage ~~containers, or in an approved shipping~~ cask.*

Technical and Regulatory Basis:

These are editorial changes to replace "irradiated fuel" with "spent fuel" and to recognize that, at LACBWR, spent fuel assemblies will only be stored in the FESW or in the approved dry spent fuel storage cask selected for use at the ISFSI. The interim time period when the spent fuel assemblies will reside temporarily in the storage canister inside the transfer cask in the cask pool during fuel loading operations is not considered "storage."

b) TS 4.1.1.2 currently states:

> *4.1.1.2 During the handling of irradiated fuel assemblies that have been operated at power levels greater than 1 Mwt, the depth of water in the Fuel Element Storage Well shall be at least 2 feet above the active fuel, and only one fuel assembly will be moved at a time.*

Proposed Change:

DPC proposes to modify this TS to read as follows (deleted text in strikethrough and new text in bold type):

> *4.1.1.2 During the handling of ~~irradiated~~ **spent** fuel assemblies that have been operated at power levels greater than 1 Mwt, the depth of water in the Fuel Element Storage Well **and the contiguous cask pool** shall be at least 2 feet above the active fuel, and only one **spent** fuel assembly will be moved at a time.*

Technical and Regulatory Basis:

The first change is an editorial change to replace "irradiated fuel assembly" and "fuel assembly" with "spent fuel assembly" in two instances. The second change extends the current TS requirement of 2 feet of water coverage over spent fuel assemblies during fuel handling within the FESW to the cask pool. During storage canister loading operations, the FESW and cask pool are made hydraulically contiguous. Spent fuel assemblies are transferred underwater from the FESW through the fuel transfer canal to the storage canister located in the cask pool. At all times while individual spent fuel assemblies are handled and transferred, at least 2 feet of water coverage will be maintained above the suspended spent fuel assembly as required by this TS. The requirement to move only one spent fuel assembly at a time remains.

c) TS 4.1.1.3 currently states:

> *4.1.1.3 With the exception of a shipping cask or transfer cask, the core spray bundle, the transfer canal shield plug and the other waste processing components and fixtures weighing less than 50 tons that are located and used within the storage well, no objects heavier than a fuel assembly shall be handled over the Fuel Element Storage Well.*

Proposed Change:

DPC proposes to completely replace this TS with the following:

> ***4.1.1.3 No object heavier than 25 tons shall be handled over spent fuel assemblies located in the Fuel Element Storage Well or cask pool. Lifting and movement of a fuel-loaded storage canister and transfer cask shall be performed using the single-failure-proof cask handling crane lifting system meeting the guidance in NUREG-0612, Section 5.1.6. Lifting and movement of objects over spent fuel assemblies located in the Fuel Element Storage Well or cask pool shall be performed in accordance with the LACBWR NUREG-0612 commitments and the dedicated project heavy load control plan.***

Technical and Regulatory Basis:

This change is proposed to clarify the weight limit for objects permitted to be suspended over spent fuel assemblies. The load limit of 25 tons provides a factor of safety of 2 for lifts performed using the Reactor Building polar crane, which has a capacity of 50 tons. Outdated language and references to obsolete or unused equipment are also removed. The use of a single-failure proof cask handling crane lifting system for cask lifts protects the spent fuel assemblies being moved while located inside the storage canister and protects the cask pool and FESW against heavy load drops beyond the plant's licensing basis. All heavy load lifts will be in accordance with DPC's commitments to NUREG-0612, such as load paths, operator training, etc., to ensure a load drop remains highly unlikely and that LACBWR's worst case analyzed accident (a 50-ton cask drop onto spent fuel assemblies) remains bounding.

LCO 4.1.2, "FUEL ELEMENT STORAGE WELL"

TS 4.1.2 currently states:

4.1.2 *FUEL ELEMENT STORAGE WELL*

LIMITING CONDITION FOR OPERATION
==

The Fuel Element Storage Well (FESW) shall meet the following requirements:

a. The Fuel Element Storage Well water level shall be at least 16 feet above any irradiated fuel stored in the spent fuel storage racks, and

b. Water in the storage well shall be maintained at a temperature ≤ 150°F.

APPLICABILITY: At all times.

ACTIONS

a. With water level less than 16 feet above any irradiated fuel stored in the Fuel Element Storage Well storage racks, take immediate action to restore water level and suspend all operations involving FUEL HANDLING.

b. With water temperature in the storage well above 150°F, take actions to reduce water temperature to ≤ 150°F within 24 hours and suspend any evolutions involving FUEL HANDLING.

SURVEILLANCE REQUIREMENTS
==

5.1.2.1 The Fuel Element Storage Well water level and FESW System water temperature shall be monitored at least once per 24 hours.

5.1.2.2 The Fuel Element Storage Well water level indication channel shall be calibrated (CHANNEL CALIBRATION) at least once per 18 months.

Proposed Changes:

DPC proposes to modify this TS as follows:

1. Modify the title of the LCO to be: "FUEL ELEMENT STORAGE WELL AND CASK POOL"

2. Add a new note preceding the LCO as follows:

"This LCO does not apply to the cask pool if the spent fuel storage canister lid is in place in the canister or if there are no spent fuel assemblies in the cask pool."

3. Modify the lead-in sentence of the LCO to add "and cask pool" after "(FESW)."

4. Modify LCO 4.1.2.a to add "and cask pool" after "Fuel Element Storage Well," change "16 feet" to "11 feet, 6½ inches," replace "irradiated fuel" with "spent fuel assembly," and add the following text after "storage racks":

   "or in a spent fuel storage canister in the cask pool, ..."

5. Modify LCO 4.1.2.b to add "and cask pool" after "storage well."

6. Modify the Applicability to state: "While spent fuel assemblies are in the FESW or the cask pool."

7. In Action 'a', replace "16 feet above any irradiated fuel stored in the Fuel Element Storage Well storage racks" with "required by the LCO."

8. Modify Action 'b' to add "or cask pool" after "storage well," and change "any evolutions" to "all operations."

9. Add a new note preceding the Surveillance Requirements as follows:

   "SR 5.1.2.1 and 5.1.2.2 do not apply to the cask pool if the spent fuel storage canister lid is in place in the canister or if there are no spent fuel assemblies in the cask pool."

10. Modify Surveillance Requirement 5.1.2.1 as follows:

    a. Add "and cask pool" after "Fuel Element Storage Well"
    b. Delete "FESW System water"
    c. Change "monitored" to "verified."
    d. Change "once per 24 hours" to "once per 12 hours."

11. Modify Surveillance Requirement 5.1.2.2 to add "and cask pool" after "Fuel Element Storage Well."

DPC proposes this TS to now read as follows (deleted text in strikethrough and new text in bold type):

*4.1.2 FUEL ELEMENT STORAGE WELL **AND CASK POOL***

LIMITING CONDITION FOR OPERATION

Note

This LCO does not apply to the cask pool if the spent fuel storage canister lid is in place in the canister or if there are no spent fuel assemblies in the cask pool.

*The Fuel Element Storage Well (FESW) **and cask pool** shall meet the following requirements:*

a. *The Fuel Element Storage Well **and cask pool** water level shall be at least ~~16~~ **11** feet, **6½ inches** above any ~~irradiated~~ **spent** fuel **assembly** stored in the spent fuel storage racks **or in a spent fuel storage canister in the cask pool**, and*

b. *Water in the storage well **and cask pool** shall be maintained at a temperature $\leq 150^{o}F$.*

APPLICABILITY: ~~*At all times.*~~ ***While spent fuel assemblies are in the FESW or the cask pool.***

ACTIONS

a. *With water level less than **required by the LCO,** ~~16 feet above any irradiated fuel stored in the Fuel Element Storage Well storage racks,~~ take immediate action to restore water level and suspend all operations involving FUEL HANDLING.*

b. *With water temperature in the storage well **or cask pool** above $150^{o}F$, take actions to reduce water temperature to $\leq 150^{o}F$ within 24 hours and suspend ~~any evolutions~~ **all operations** involving FUEL HANDLING.*

SURVEILLANCE REQUIREMENTS

Note

SR 5.1.2.1 and 5.1.2.2 do not apply to the cask pool if the spent fuel storage canister lid is in place in the canister or if there are no spent fuel assemblies in the cask pool.

*5.1.2.1 The Fuel Element Storage Well **and cask pool** water level and ~~FESW System water~~ temperature shall be ~~monitored~~ **verified** at least once per ~~24~~ **12** hours.*

*5.1.2.2 The Fuel Element Storage Well **and cask pool** water level indication channel shall be calibrated (CHANNEL CALIBRATION) at least once per 18 months.*

Technical and Regulatory Basis:

These proposed changes, on the whole, are to reflect the inclusion of the cask pool within the scope of the LCO, to change the 16 feet water coverage to 11 feet, 6½ inches water coverage, to remove the previously proposed new LCO 4.1.2.c, and to make editorial improvements. The justification for each change is provided below consistent with the numbering sequence above.

1. This is an editorial change to clarify the scope of the LCO as being applicable to the FESW and the cask pool.

2. This new note is added to recognize that there are no minimum water coverage requirements over spent fuel assemblies in the spent fuel storage canister after the canister lid has been installed or if there are no spent fuel assemblies in the cask pool. The canister lid is a thick shield lid designed and licensed for use in accordance with 10 CFR 72 to permit the fuel-loaded canister to be removed from the spent fuel pool and prepared for storage operations at the ISFSI in accordance with the dry storage cask CoC and FSAR (i.e., lid welding and canister draining, drying and helium backfilling).

   In order to remove the canister and transfer cask from the cask pool, the cask pool gate must be removed. In order to remove the cask pool gate, the water level in the cask pool must be lowered to an elevation below the bottom of the cask pool gate, or approximately elevation 678 feet. This elevation is below the top of the transfer cask and canister. Without this note, the LCO requirements would prevent lowering the cask pool level to a point sufficient to remove the cask pool gate. The shielding provided by the water above an open canister (required by the LCO) is effectively replaced by the shielding provided by the canister lid, making the LCO unnecessary after the lid is in place.

   If there are no spent fuel assemblies in the cask pool, there is no need for water shielding or temperature monitoring.

3. This is an editorial change to clarify the scope of the LCO as being applicable to the FESW and the cask pool.

4. This change proposes to reduce the minimum required water level over spent fuel assemblies in the FESW and cask pool to 11 feet, 6½ inches. The elevation of the top of active fuel in the upper and lower tier storage racks and the minimum water level elevation required to comply with the proposed TS are:

LOCATION	Top of Active Fuel Elevation	TS Minimum Water Elevation
FESW Upper Tier	677'-2⅛"	688'-8⅝"
FESW Lower Tier	668'-2½"	679'-9"
Cask Pool	678'-6"	690'-½"*

* With spent fuel in the canister and the canister closure lid not in place

The value of 11 feet, 6½ inches was chosen because it provides adequate shielding of the spent fuel assemblies and represents the minimum required level of water to prevent vortexing of air into the FESW pump suction piping. Both of these issues are discussed in more detail below.

Shielding

The minimum proposed level of water over spent fuel assemblies is 11 feet, 6½ inches. LACBWR Special Test Procedure, STP-58-01, "Perform Radiation Survey in the FESW," (Reference 4) was performed on April 27, 2010, to measure the dose rates in the FESW above the upper tier spent fuel racks. A Thermo-Scientific FH 40 GX Survey Meter, fitted with a model FHZ 312, 20-meter underwater detector having a range of 10 mRem/hr to 10,000 Rem/hr, was used to take the dose rate readings. Dose rates were measured at nine survey points in a 3x3 array extending over the entire FESW at 10 feet, 8 feet, 6 feet, and 4 feet above the upper tier racks. All readings at 10 feet and 8 feet above the upper tier racks were measured as less than 10 mRem/hr. At 6 feet above the upper tier racks one reading at Survey Point #2 measured 11.1 mRem/hr with the other eight points measuring less than 10 mRem/hr. At 4 feet above the upper tier racks five of nine points measured in a range from 22.3 mRem/hr to 11.9 mRem/hr with the other four points measuring less than 10 mRem/hr.

The instrument was checked at a known 13 Rem/hr exposure rate point in the FESW before and after each survey with no appreciable variation in readings during each check demonstrating that the instrument was functioning properly.

It should be recognized that the measured dose rates were inside the FESW at approximately elevation 688, 686, 684, and 682 feet. The requested minimum water coverage of 11 feet, 6½ inches equates to water level at elevation 688 feet, 8⅝ inches. Current dose rates with FESW water level maintained at elevation 695-700 feet are 1 mRem/hr or less. With all but one measured dose rate dropping off to less than 10 mRem/hr at 6 feet above the upper tier racks (elevation 684 feet), additional water coverage to elevation 688 feet, 8⅝ inches minimum with spent fuel assemblies in the upper tier racks assures that dose rates to personnel located on the fuel handling floor (elevation 701 feet) will be no different than they are with the FESW full of water.

Vortexing

Sargent & Lundy provided analysis of the vortexing issue; this analysis is documented in Reference 5. The minimum submergence required to prevent vortexing is calculated based on the Froude number in the equation below:

$$\frac{S}{D} = 1.0 + 2.3 \times Fr_D$$

Where: S – Submergence to the center of the pipe (ft)
D – Pipe inner diameter (ft)

Fr_D – Froude number $Fr_D = \frac{V}{\sqrt{g \times D}}$
V – Entrance velocity of the pipe (ft/s)
g – Acceleration due to gravity (32.2 ft/sec^2)

The equation can also be expressed as the following:

$$S_{min} = D + 0.574\left(\frac{Q}{D^{1.5}}\right)$$

Where: S_{min} – Minimum submergence to the center of the pipe (in)
D – Pipe inner diameter (in)
Q – Volume flow rate (gpm)

The Schedule 40 pipe inner diameter is 0.505 feet (6.065 inches). The FESW cooling system normal flow rate is 150 gpm and can be throttled to 75 gpm.

For 150-gpm FESW cooling system flow rate, the minimum submergence is:

$$S_{min} = 6.065 + 0.574\left(\frac{150}{6.065^{1.5}}\right) = 11.829 \text{ inches}$$

For 75-gpm FESW cooling system flow rate, the minimum submergence is:

$$S_{min} = 6.065 + 0.574\left(\frac{75}{6.065^{1.5}}\right) = 8.497 \text{ inches}$$

To prevent vortexing, minimum submergence of the 6-inch suction line with a FESW cooling system flow rate of 150 gpm is approximately one foot of water above pipe centerline elevation 679 feet. FESW water level will generally be kept at between elevation 680 feet and 680 feet, 5 inches during FESW cooling system operation at low water level (i.e., below 680'-5"). The FESW cooling system flow rate can be controlled by throttle valves to approximately 75 gpm with little impact to system cooling or pump operation. The resulting minimum submergence to prevent vortexing at 75 gpm is approximately 9 inches, or water level elevation 679'-9".

5. This is an editorial change to clarify the scope of the LCO as being applicable to the FESW and the cask pool.

6. This is an editorial change to clarify the scope of the LCO as being applicable to the FESW and the cask pool.

7. This is an editorial improvement to simply refer to the LCO not being met rather than repeating the LCO.

8. This is an editorial change to clarify the scope of the LCO as being applicable to the FESW and the cask pool and to make the language in the two action items consistent.

9 This new note is added to recognize that there are no minimum water coverage requirements over the spent fuel assemblies in the spent fuel storage canister after the canister lid has been installed or if there are no spent fuel assemblies in the cask pool. Thus, the Surveillance Requirements do not need to be performed for the cask pool under those circumstances. See also the justification for Change 2 above.

10. Change 10a is an editorial change to clarify the scope of the SR as being applicable to the FESW and the cask pool. Change 10b is a request to clarify and simplify the SR to require verification of the FESW and cask pool water level <u>and</u> temperature. Change 10c is an editorial improvement that is more common terminology consistent with Surveillance Requirements that require specific limits to be confirmed to be met. Change 10d increases the frequency of water level and temperature monitoring from every 24 hours to every 12 hours. Rather than keep the 24-hour frequency and shift to a 12-hour frequency at lowered level, DPC has determined that, from a human factors standpoint, it was prudent to change the surveillance frequency at all times to every 12 hours. Because the minimum level over spent fuel assemblies is now proposed to be 11 feet, 6½ inches rather than 2 feet 9 inches, it was considered reasonable to modify the previous proposed frequency of 6 hours to 12 hours. A 12-hour frequency also aligns with the 12-hour operating shifts employed at LACBWR.

11. Change 11 is an editorial change to clarify the scope of the SR as being applicable to the FESW and the cask pool.

<u>RAI 2</u>

The amendment request proposes changes to TS requirement 4.1.2 for the minimum depth of water in the FESW, from 16 feet above irradiated fuel in the fuel storage racks, to: (1) 10 feet [referring to the depth of water above fuel in the *lower* tier storage rack when the FESW is drained to its lowest level (approximately 680 foot elevation), to allow for cask movement and preparation]; and (2) no less than two feet, nine inches (referring to the depth of water above fuel in *upper* tier storage rack when the FESW is drained to its lowest level at approximately 680 foot elevation).

LACBWR TS 4.1.1.2 requires a depth of at least two feet of water above a fuel assembly during fuel handling, and the LAR states that the water level will be maintained at the approximately 695 foot elevation while fuel is being moved through the fuel transfer canal to the cask pool, to meet this TS requirement. It is not clear from the license amendment request how the water level will fluctuate in the cask pool during the loading operations.

Please provide additional details regarding the cask loading operations overview, specifically addressing changes in cask pool water levels as water is added and drained from the FESW and cask pool. Please specifically address water levels in the cask pool at the time the canister lid is installed.

Response to RAI 2

DPC is rescinding its request to permit FESW water level to be as low as 2 feet, 9 inches above the spent fuel assemblies in the storage racks. LCO 4.1.2 is now extended to apply to the FESW and the cask pool, and limits the water coverage over spent fuel assemblies to a minimum of 11 feet, 6½ inches as described in the response to RAI 1. The operational sequences from initial conditions until the transfer cask and storage canister are removed from the cask pool, including water levels, gate operations, cask handling crane girder status, and lifting hoists being used, both with and without spent fuel assemblies in the upper tier FESW storage racks, are shown in the tables below (refer to the Sketch S-09-0001, FESW and Cask Pool Layout, provided as Attachment 2 to the original LAR submittal):

With Fuel in the Upper Tier FESW Spent Fuel Racks – Dry Runs and First Two Casks
(all water level elevations are approximate)

ACTIVITY*	CASK POOL		FESW		CHC GIRDERS	LIFTING DEVICE USED
	Gate	Water Level (ft. elevation)	Canal Gate	Water Level (ft. elevation)		
Initial conditions	Out	≤ 678	In	695 – 700	Installed	NA
Place TC/TSC into cask pool	Out	≤ 678	In	695 – 700	Installed	Cask Handling Crane (CHC)**
Park CHC trolley outside	Out	≤ 678	In	695 – 700	Installed	NA
Stage CHC girders	Out	≤ 678	In	695 – 700	Removed	Polar Crane (PC)
Install cask pool gate	In	≤ 678	In	695 – 700	Removed	PC
Fill cask pool with water	In	695	In	695	Removed	NA
Remove and stage transfer canal gate	In	695	Out	695	Removed	PC
Load TSC with fuel	In	695	Out	695	Removed	Fuel Bridge
Install TSC closure lid	In	695	Out	695	Removed	PC
Re-install transfer canal gate	In	695	In	695	Removed	PC
Lower cask pool water level	In	≤ 678	In	695 – 700	Removed	NA
Remove cask pool gate	Out	≤ 678	In	695 – 700	Removed	PC
Re-install CHC girders	Out	≤ 678	In	695 – 700	Installed	PC
Move TC/TSC out of cask pool to cask prep area	Out	≤ 678	In	695 – 700	Installed	CHC

* TC = NAC-MPC Transfer Cask; TSC = NAC-MPC Transportable Storage Canister
** Single-failure-proof lifting system per NUREG-0612, Section 5.1.6

Without Fuel in the Upper Tier FESW Spent Fuel Racks – Final Three Casks
(all water level elevations are approximate)

ACTIVITY*	CASK POOL		FESW		CHC GIRDERS	LIFTING DEVICE USED
	Gate	Water Level (ft. elevation)	Canal Gate	Water Level (ft. elevation)		
Initial conditions	Out	≤ 678	In	695 – 700	Installed	NA
Place TC/TSC into cask pool	Out	≤ 678	In	695 – 700	Installed	CHC**
Park CHC trolley outside	Out	≤ 678	In	695 – 700	Installed	NA
Stage CHC girders	Out	≤ 678	In	695 – 700	Removed	PC
Install cask pool gate	In	≤ 678	In	695 – 700	Removed	PC
Fill cask pool with water	In	695	In	695	Removed	NA
Remove and stage transfer canal gate	In	695	Out	695	Removed	PC
Load TSC with fuel	In	695	Out	695	Removed	Fuel Bridge
Install TSC closure lid	In	695	Out	695	Removed	PC
Lower cask pool and FESW water level	In	≤ 678	Out	≥ 679'-9"	Removed	NA
Remove cask pool gate	Out	≤ 678	Out	≥ 679'-9"	Removed	PC
Re-install CHC girders	Out	≤ 678	Out	≥ 679'-9"	Installed	PC
Move TC/TSC out of cask pool to cask prep area	Out	≤ 678	Out	≥ 679'-9"	Installed	CHC

* TC = NAC-MPC Transfer Cask; TSC = NAC-MPC Transportable Storage Canister
** Single-failure-proof lifting system per NUREG-0612, Section 5.1.6

RAI 3

TS 2.2.3 requires that the FESW shall be designed and maintained to prevent inadvertent draining of the FESW below an elevation of 679 feet.

The LAR proposes to drain the connected cask pool area and FESW in order to remove the cask pool gate to remove filled casks from and place empty casks into the cask pool. The amendment request does not identify the point of drainage from the cask pool.

a. Please describe the method by which the cask pool will be drained and the point from which this drainage will be taken.

b. Please identify any physical aspects of the drainage flowpath that would limit accidental drainage from the cask pool and the FESW.

Response to RAI 3a

Refer to sketch S-09-0001, FESW and Cask Pool Layout, provided as Attachment 2 to the original LAR submittal.

The cask pool water level must be lowered to approximately elevation 678 feet once the storage canister lid has been placed into position in the storage canister in order to remove the cask pool gate. There are two possible final configurations of the cask pool and the FESW depending on the status of the fuel transfer canal gate. With the transfer canal gate installed, only the cask pool water level will need to be lowered. With the transfer canal gate removed, both the FESW and the cask pool water level will need to be lowered to the bottom elevation of the transfer canal (680'-5"). The cask pool water level will then continue to be lowered to approximately elevation 678 feet. The FESW water level will be permitted to be lowered to elevation 679'-9" per proposed revised LCO 4.1.2.

The cask pool below the elevation of the bottom of the cask pool gate at approximately elevation 678 feet (also known as the "lower cask pool") has two penetrations in the pool wall, each with double isolation valves. The lower penetration is near the bottom of the pool at approximately elevation 669'-3" and the upper penetration is at approximately elevation 676'-9". During the storage canister loading evolution, a water clean-up system will be connected to these lower cask pool penetrations. The primary purpose of the water clean-up system is to provide a source of clean water to the transfer cask/storage canister annulus to ensure that the surface of the canister does not become contaminated by cask pool water above TS limits for the NAC-MPC System. The water clean-up system will also be used to raise or lower cask pool water level when the cask pool is separated from the FESW.

Two operating scenarios, with and without the fuel transfer canal gate installed are discussed below.

I. Fuel Transfer Canal Gate Installed

The method that will be used to lower the water level in the cask pool to permit removal of the cask pool gate when the fuel transfer canal gate is installed is as follows:

1. After storage canister fuel loading is complete, the canister closure lid is placed into the canister while the water level is maintained at approximately elevation 695 feet.

2. The fuel transfer canal gate is installed to separate the cask pool from the FESW.

3. Using the cask pool water clean-up system, the water level in the cask pool is lowered. FESW water level is controlled between elevations 695 and 700 feet.

4. The cask pool water level is lowered until the transfer cask trunnions are exposed and the water level clears the bottom of the cask pool gate at approximately elevation 678 feet.

When the lowering of cask pool water level is complete, the FESW water level will be controlled between approximately elevations 695 and 700 feet. The cask pool water level will be at approximately 678 feet. Both the loaded storage canister and the annulus between the transfer cask and the storage canister will remain full of water, with the storage canister lid in place.

II. Fuel Transfer Canal Gate Removed

The method that will be used to lower the water level in the FESW and the cask pool when the fuel transfer canal gate is not installed is as follows:

1. After storage canister fuel loading is complete, the canister closure lid is placed into the canister while the water level is maintained at approximately elevation 695 feet.

2. Using the FESW cooling system pumps, the water level in both the cask pool and FESW is lowered to the bottom of the transfer canal at elevation 680 feet 5 inches. The cask pool water level is lowered further using the water clean-up system.

3. The cask pool water level is lowered until the transfer cask trunnions are exposed and the water level clears the bottom of the cask pool gate at approximately elevation 678 feet.

When the lowering of cask pool and FESW water levels is complete, the FESW water level will be controlled between elevations 679 feet, 9 inches and 680 feet, 5 inches with about 12 feet of water above the active fuel of the spent fuel assemblies in the FESW lower tier racks. The cask pool water level will be at approximately 678 feet. Both the loaded storage canister and the annulus between the transfer cask and the storage canister will remain full of water, with the storage canister lid in place.

Response to RAI 3b

The removal of water from the cask pool and the FESW will be controlled by a written procedure detailing the valve line-ups and pumping methods. The FESW cooling system pumps are operated from the control room where the operator has two separate level indications to monitor the level in the FESW. The cask pool water clean-up system is operated locally.

The FESW is physically protected from accidental draining by the north wall of the FESW, the fuel transfer canal gate (if installed) and the location of the FESW pump suction line at the 679 feet elevation. All FESW penetrations below this elevation have been capped. As a result, the FESW water level is prevented from inadvertently draining below 679 feet or approximately 2 feet above the top of the spent fuel assemblies stored in the upper fuel storage racks.

The cask pool is physically protected from accidental draining by the installation of the cask pool gate, the ability to monitor the gate seal for leakage, and double isolation valves on both of the penetrations into the lower cask pool. The cask pool gate can be monitored for leakage and the penetration valves are accessible to be closed from the mezzanine level in the Reactor Building in the event of a leak in the piping.

Once the transfer cask containing a spent fuel storage canister is placed in the cask pool and the pool is flooded to start cask loading operations, the canister will remain full of water irrespective of the water level in the cask pool around it because there are no penetrations in the storage canister that could allow drainage.

RAI 4

Surveillance requirements in TS 5.1.2.1 currently require the licensee to monitor the FESW water level and water temperature once per 24 hours. The amendment request proposes a change to this TS to require a more frequent surveillance and verification of FESW water level when FESW water level is lower than 10 feet above irradiated fuel, to allow more time to take corrective actions if the FESW cooling system is affected. Specifically, the licensee is proposing to increase the frequency of verification of the water *level* to once per 6 hours. The proposed TS change does not include an increase in frequency of verification of the water *temperature*.

a. Please explain why an increased frequency of verification of the water temperature, during lower water level conditions (when FESW water level is lower than 10 feet above the fuel), is not necessary.

b. Please explain whether the water level and temperature are monitored continuously and whether there are alarm set points regarding water levels and temperatures.

c. Please explain the effects of any vortexing or air entrapment for the FESW pump suction piping located at the 679 foot elevation of the FESW, if the water were to inadvertently drain to this level.

Response to RAI 4a

As described in the response to RAI 1, DPC has decided to simplify the proposed LCO change to request a reduction in the minimum required water coverage over spent fuel assemblies from 16 feet to 11 feet 6½ inches. Consistent with that change, the frequency for Surveillance Requirement 5.1.2.1 is proposed to be reduced from every 24 hours to every 12 hours for both water level and temperature. Verification of water level and temperature every 12 hours was chosen as a reasonable increase in the frequency that is commensurate with the proposed level change and it coincides with the length of an operating shift at LACBWR.

Response to RAI 4b

FESW water level and temperature are continuously monitored and alarmed. Two level transmitters provide two diverse FESW water level indications and one alarm function. The primary FESW water level transmitter supplies an indicator on the control room main control panel. The level transmitter signal actuates the "Fuel Element Storage Well Level (LO) (HI)" alarm in the control room when FESW water level decreases to elevation 695 feet, or increases to elevation 700 feet.

The secondary level indication is from a bubbler-type level transmitter. This transmitter supplies a computer display in the control room. This level indicator does not include an audible control panel alarm. However, it does provide an alarm on the computer screen in the control room as a diverse means for an operator to check the validity of a control panel alarm from the primary indication. Iron constantan thermocouples are installed in the FESW cooler inlet piping and in the pipe length downstream of the cooler outlet and cooler bypass lines. These thermocouples provide remote indication of FESW temperature and actuate a high temperature alarm in the control room at 135°F. Installed in the piping near the thermocouples are 0-200°F gauges for local water temperature indication. Should the FESW cooling system not be operating, FESW/cask pool temperature will be monitored using portable temperature monitoring equipment.

Response to RAI 4c

During operation of the FESW cooling system pumps at lower FESW levels, suction will be from the main suction line with a centerline elevation of 679'-0". The main suction line is a nominal 6 inch diameter, Schedule 40 pipe. FESW water level will be maintained between elevations 679'-9" and 680'-5", which provides a minimum of 9 inches of submergence above the centerline of the pipe. As discussed in the Technical and Regulatory Basis for changes at TS 4.1.2, calculations show that the minimum level of coverage to prevent vortexing at an FESW pump flow rate of 75 gpm in a 6-inch diameter, Schedule 40 suction line is 9 inches above the pipe centerline, or 679'-9". When required, system flow rate will be controlled at approximately 75 gpm using the throttle valves at the FESW cooler exit and in the cooler bypass line.

The two FESW cooling system pumps are centrifugal pumps, each of which can provide adequate flow for spent fuel pool cooling. Normally one pump operates at a time. A centrifugal pump can withstand a minor amount of air entrainment with no deleterious effect on pump operation. If the amount of air entrainment becomes significant, the pump would display flow oscillations. If this were to happen, the operators would recognize the erratic flow on the system flow instrumentation and secure the operating pump. Any trapped air would have to be vented from the pump. Once adequate level was re-established, an FESW pump would be started to restore cooling flow.

Based on the Loss of FESW Cooling accident described in the LACBWR Decommissioning Plan (D-Plan), Section 9.4, the FESW pool temperature after a full loss of FESW cooling rises to about 114°F in approximately 15 days (In year 1993 heat load – about 12 kW total). Extrapolation of the data shows that the pool temperature would stabilize at approximately 150°F. This analysis is bounding for today's heat load, which remains approximately the same, and demonstrates that there is more than adequate time to recover the pump.

RAI 5

The LAR states that the geometry and poison (boron) loading of the fuel storage rack cells ensures the FESW is maintained subcritical and keff (effective multiplication factor) in the FESW does not exceed 0.95 at any time. LACBWR Operations Procedure OP-58-02, "Irradiated Fuel Element Storage Rack Poison Material Surveillance Program," is used at LACBWR to monitor the long term performance of and verify the integrity of the neutron absorber material in the spent fuel storage racks, which is required to control the reactivity of the fuel storage system. The licensee tests the loss of boron in the neutron absorbers by measuring the weight of the surveillance coupons.

Since the boron loading is credited to maintain the subcriticality of the pool, the staff has questions regarding the material condition of the neutron absorbing material in the storage racks. In order to have reasonable assurance that the neutron absorbing material will be able to perform its intended function during movement of fuel assemblies and work in the FESW during cask loading operations, the staff requests the following information.

a. In OP-58-02, a weight loss of 10 percent or less for any sample (of the composite poison material) is acceptable, and no further examinations of these or other samples are required. Please clarify the rationale for the upper limit of a 10 percent weight loss. Also, please state what boron-10 areal density a 10 percent weight loss correlates to and discuss why it is justified.

b. Please discuss whether neutron attenuation testing has been performed on the coupons/racks. If so, please discuss the dates of the tests performed and the results.

c. OP-58-02 indicates that some of the coupons were sent to Northeast Technology Corp (NETCO) for testing in 1997. Please discuss why the coupons were sent to NETCO for testing, what testing was performed, and the results of the testing.

d. OP-58-02 states that visual testing is also performed on the coupons. Please provide and discuss the results of these tests.

e. Please provide and discuss trending data of the coupons.

f. Please discuss what calculations are performed to determine that keff ≤ 0.95 is maintained, based on the results of the surveillance in OP-58-02.

<u>Response to RAI 5</u>

This response is provided first with a general response below and individual responses to each of the separate sub-questions thereafter.

The LACBWR fuel storage rack design includes neutron absorbing material (poison material) between any two adjacent spent fuel assemblies. This neutron absorber material, supplied by the Carborundum Company, is a composite of a phenol-formaldehyde polymer (~18 w/o) and B_4C particles (~64 w/o) bonded to a woven glass-fiber reinforcement (~18 w/o) by the same polymer. This poison material is built into the walls of the fuel storage cells and is protected from mechanical effects of fuel handling and cooling water flow, etc. by stainless steel cover plates tack welded to the cell walls every few inches on all sides. The poison material spaces are vented to the FESW environment.

For surveillance purposes DPC provided a neutron poison coupon sample holder that positioned samples of the Carborundum poison material next to spent fuel assemblies in the storage racks in the approximate geometry of the poison material built into the racks. The coupons on this sample holder have experienced a more rigorous environment than the poison material in the racks for several reasons:

1. The coupons on the sample holder are covered with stainless steel covers, closed on only two sides instead of being completely covered by cover plates like the poison material in the racks. Visual examination of the sample coupons show that the coupons are significantly affected along the two exposed edges compared to the covered edges. This is attributed to FESW cooling water flow-induced erosion, to which the neutron absorber in the racks would not be subjected.

2. The coupon sample holder is dimensionally very similar to a spent fuel assembly and is handled with the same equipment. Therefore it has been used for years as a dummy fuel assembly for fuel handler training. During refueling outages the coupon sample holder was repositioned next to newly discharged, high exposure spent fuel assemblies so that the neutron and gamma dose to the coupons would be maximized. The coupons that have been examined under LACBWR Operations Procedure OP-58-02 have been removed from the sample holder, (some as many as four times over the years), handled, placed in plastic bags for transport to laboratories, dried, photographed, visually examined, weighed, transported back to the plant, and replaced on the sample holder. The poison material in the racks has not experienced the stresses described above since it has been protected by the tack welded cover plates since original rack fabrication.

3. The poison material specification for the LACBWR fuel racks required that the minimum areal density of the B^{10} be 0.024 gm/cm^2 after simultaneous exposure to demineralized water and 10^{11} Rad gamma irradiation. The material supplied by Carborundum did not meet this specification so it was decided that two sheets of the Carborundum material would be used in each poison region of LACBWR racks. With two sheets of the composite poison material in each poison region of the racks, the as-built areal density of the B^{10} is nominally 0.038 gm/cm^2. Current exposure to the poison material in the FESW is approximately 10^{10} Rad.

Response to RAI 5a

The original poison material is 64 w/o B_4C as stated above. Conservatively assuming that the 10 w/o of the original material weight lost is all B_4C, the weight of the B_4C remaining is 54 w/o of the original weight. Therefore, the fraction of original B_4C remaining is 54/64 = 0.84375 and the areal density of the B^{10} is 0.84375(0.038) = 0.0321 gm/cm^2 or 1.337 times the design requirement of 0.024 gm/cm^2.

Additional conservatism was provided in the criticality analyses of the LACBWR fuel storage racks which assumed that the nominal areal density of B^{10} in the Carborundum neutron absorbing material in the racks was 0.022 gm/cm^2. Using this B^{10} areal density and taking into account normal as-built variations, the "worst case abnormal configuration, and no burn-up credit, the upper limit for k_{eff} was calculated to be 0.9275. When a B^{10} areal density of 0.0195 gm/cm^2 was assumed, the worst case upper limit k_{eff} was calculated to be 0.9325. The regulatory limit for spent fuel racks is $k_{eff} \leq 0.95$.

From the above discussion, it is apparent that the LACBWR fuel racks would meet criticality requirements after a uniform loss of up to 50 w/o of original B^{10} in the poison regions of the racks. Therefore, the surveillance procedure acceptance criteria of less than or equal to 10% sample weight loss along with visual confirmation of the general physical integrity of the samples is conservative.

Response to RAI 5b

Before the spent fuel storage racks were installed, each location in the racks designed to contain Carborundum poison material was visually examined to assure the presence of material and a neutron attenuation test was performed to assure the presence of a strong neutron absorber in each location. This test did not measure B^{10} areal density. All locations contained neutron absorber material.

No neutron attenuation testing has been performed on the poison material in the installed fuel storage racks. See the answer to RAI 5c below for the results of neutron attenuation testing of two surveillance coupons performed by Northeast Technology Corporation (NETCO) in 1997. The surveillance coupons placed on the sample holder were archive samples of material used in the racks.

Response to RAI 5c

In order to obtain additional assurance of the integrity of the poison material in the LACBWR spent fuel storage racks, the two samples (5B Inner and 5B Outer) removed from position 5B on the sample holder in March 1997 were sent to NETCO for examination and measurement of Boron-10 areal density. These samples had been exposed to near maximum gamma fluxes in the fuel racks for 16.3 years. Measurements at LACBWR before shipment to NETCO indicated weight loss of 4.37% for 5B Inner and 4.70% for 5B Outer. One unirradiated archival coupon was also sent to NETCO.

The following tests were performed on each coupon:

- Visual inspection and high resolution photography
- Coupon dimensions
- Dry weight
- Specific gravity and density
- Radioassay
- Neutron attenuation testing

The physical examinations at NETCO substantiated the observations at LACBWR, that a significant part of the weight loss could be attributed to water erosion of the unprotected sample edges in the sample holder geometry and to physical damage during handling of the samples during the current and three previous examinations. The poison material in the fuel racks is not subjected to water erosion or physical damage since it is essentially sealed under SS cover plates. The Boron-10 areal density determined at NETCO by neutron attenuation measurements was 0.0211 gm/cm^2 for 5B Inner, 0.0193 gm/cm^2 for 5B Outer, and 0.0207 gm/cm^2 for the archive coupon. Since two layers of the poison material are always used together, the resultant Boron-10 areal density for the 5B location on the sample holder is 0.0404 gm/cm^2. This is more than twice as much as required by the criticality analysis.

Response to RAI 5d

After the coupons were transported to the laboratory and air dried for at least 24 hours, they were carefully visually examined to obtain an indication of their physical integrity and condition compared to unused archival poison material. The overall condition of all coupons examined has been quite good. The surface appearance in general is very similar to the archive material. The bulk of the B_4C matrix material tightly adheres to the fiberglass backing material. No major cracks or missing chips have been observed. The surface of the material is a little more friable than the archive material and fine particles will rub off when the material is handled as evidenced by a few particles left in the bags the material is transferred in and on white gloves the material was handled with. The surveillance coupons on the sample holder are covered by cover plates essentially closed at the top and bottom of the coupons, but the sides of the coupons are exposed to the flow of cooling water and to relative water motion caused by movement of the sample holder for fuel handler training, etc. Along the exposed edges of each of the coupons, significant erosion of the B_4C matrix is evident with up to 1/16-inch of fiberglass backing material completely exposed on some of the coupons. The water stains and appearance of the material indicates that the thinning of the matrix extends some distance in from the exposed edge.

The indications of material deterioration and loss discussed above are not expected to be significant in the poison material in the storage racks, since in the rack design the material is essentially sealed under stainless steel cover plates that are tack-welded every few inches on all sides. Probably much of the weight loss observed in the surveillance coupons is due to the unique conditions experienced by the coupons and would not occur in the actual storage rack poison material.

Response to RAI 5e

The attached spreadsheet presents the weight loss data for all surveillance coupons weighed and examined since the coupon surveillance program was initiated in October 1980 right after the present fuel storage racks were installed in the LACBWR Fuel Element Storage Well. The data indicate that the poison material is quite stable in the LACBWR FESW environment. The higher weight loss values indicated by the coupons from the 5D and 6A locations on the sample holder measured in April 2005 are not fully understood and appear to be anomalies in the data set. Even though they are a little greater than the conservative acceptance criteria in the procedure, the values are acceptable from a criticality control point of view as explained in the other responses to this RAI. Also, a second set of coupons obtained at that time from locations 7A and 7B on the coupon sample holder yielded values that were within the procedure acceptance criteria.

The data collected under the LACBWR fuel storage rack poison material surveillance program over the last 29 years and presented in the spreadsheet indicate that there is no concern that the neutron absorbers have deteriorated to the point that the design function of the storage racks in their role to prevent an inadvertent criticality is significantly affected.

Response to RAI 5f

Please refer to the response to RAI 5a.

RAI 6

The licensee used a special test procedure STP-58-01, "Perform Radiation Survey of FESW at Canal Gate Level," to obtain underwater radiation measurements needed to calculate the dose rate at the 701 foot (701') elevation in the reactor building. This information is needed to assess potential occupational doses for the proposed operations.

STP-58-01 does not provide any information (e.g., make or model number) on the radiation survey instrumentation used. STP-58-01 refers to another procedure, HSP-02.6 "Radiation Surveys," which may contain this information, but this procedure was not provided for NRC staff review.

Please provide the technical specifications provided by the radiation instrumentation manufacturer or provide a description of the design and capability of the proposed radiation instrumentation used with STP-58-01.

Response to RAI 6

The detector used to measure dose rates in the FESW is a Thermo Scientific underwater detection system (UWDS) detector, model FHZ312 with an FH40 GX display unit. This UWDS is a high-range (10 mRem/hr to 10,000 Rem/hr), energy-compensated Geiger-Mueller (GM) tube-based detector designed to be continuously used in areas where high exposure levels exist. The detector is watertight to a depth of 20 meters and the detector housing and cable are waterproof. The unit was originally calibrated by the manufacturer. Re-calibration cannot be performed offsite because the unit is contaminated after use at LACBWR. However, an instrument check is performed onsite using a point of known exposure rate in the FESW to verify that the detector readings are within the expected range. This check is performed before and after each use.

RAI 7

The licensee provided a July 1, 2009, memorandum, which described a calculation of the dose rate at the 701' elevation using the 680' 5" elevation radiation survey measurement. The exposure rate measured at the 680' 5" elevation was 1030 milliroentgen/hour (mR/hr) or an assumed dose rate of 1030 millirem/hour (mrem/hr). The licensee used this measured dose rate to calculate a dose rate at the 701' elevation of 25.3 mrem/hr.

The equation that the licensee used in the calculation of the 701' elevation dose rate appears to be incorrect. The equation contains an additional $1/r^2$ term in the denominator, and this term does not appear to be necessary since the "$\ln (r^2 + d^2) / d^2$" term accounts for the geometrical attenuation of the dose rate. Also, the dose rate equation for a disk source should contain an addition multiplicative term of pi (π) in the numerator.

NRC staff used the equation for a disk source dose rate, $I = \pi I_o \ln [(r^2 + d^2) / d^2]$ from Herman Cember & Thomas Johnson, 4th Edition, *Introduction to Health Physics*. This equation for the disk source dose rate is also used in *Principles of Radiation Shielding* by Arthur Chilton et al. Using this equation and the licensee's radiation measurement of 1030 mrem/hr, the NRC staff calculated a dose rate for the 701' elevation of 225 mrem/hr.

NRC staff also used the Microshield computer code to determine the dose rate and used a simple model of a 3-meter diameter x 3-meter height cylindrical radiation source. NRC staff used cesium-137 and cobalt-60 (separately) as the radioactive material in the source, and the source concentration was adjusted to provide a dose rate of 1030 mrem/hr at the 680' 5" elevation. Then, the dose rate was calculated at the 701' elevation using Microshield. The dose rates calculated were in the 200 – 300 mrem/hr range. The staff's results for the Microshield calculations and the calculation using the disk source equation appear to be consistent.

Please provide an explanation for the apparent discrepancy between dose rate calculations provided in the LAR and NRC staff calculations, or provide a revised dose rate calculation.

Response to RAI 7

Referring to our response to RAI 1, DPC has decided to modify our request to change the water coverage required over spent fuel assemblies in both the upper and lower tier storage racks to a minimum of 11 feet, 6½ inches. Therefore, the estimated dose rate at elevation 701 feet with spent fuel assemblies in the upper tier racks and only 2 feet 9 inches of water coverage is no longer used.

DPC has reviewed the method used to estimate the dose rate in the original LAR submittal and we feel the equation was not the most accurate approach to extrapolating dose rate from the type of source represented by the storage racks and the stored spent fuel assemblies in the FESW. A corrective action report has been generated internally to address the use of the equation.

DPC requested the assistance of Sargent & Lundy (S&L) to provide a third-party opinion on the matter. S&L provided an informal estimation of dose rates based on the previous highest survey result of 1030 mrem/hr at the 680'-5" elevation. S&L developed several models using MicroShield. Their results were in-between the DPC 25 mrem/hr and the NRC 225 mrem/hr.

As discussed in other sections of our response, minimum water coverage of 11 feet, 6½ inches gains for us a great deal of operational flexibility. The more conservative minimum water coverage reduces the uncertainty of actual dose rates we can expect to contend with and provides a greater degree of radiation safety for our fuel loading campaign consistent with good ALARA practices. The dose rates that were measured in the FESW on April 27, 2010, discussed previously in our response, give credence to our position that dose rates during periods at reduced water level will not be significant nor adversely affect our operations. These detailed FESW surveys are being submitted in lieu of a revised dose rate calculation to show little to no effect on the dose rates at elevation 701 feet with a minimum of 11 feet, 6½ inches water coverage above the stored spent fuel assemblies.

RAI 8

The licensee provided a single dose rate calculation in its July 1, 2009, memorandum as the highest dose rate anticipated at the 701' elevation, when the FESW is drained to its lowest level (approximately 680 foot elevation) to allow for cask movement and preparation. NRC staff does not consider a single dose rate calculation to adequately represent the potential worker doses for the operations proposed. The licensee did not provide adequate information on operations (e.g., change in water levels, and thus the shielding provided, during each phase of the operation and the estimated time for each phase of the operation), to determine anticipated occupational doses that may be accrued during the proposed operations.

The licensee also did not provide adequate information to support its claim that the proposed fuel loading operations per this license amendment request (to lower water levels in the FESW, allowing the fuel transfer canal gate to be removed at the onset of the dry run and not be reinstalled for the remainder of the cask loading operations) will result in lower occupational doses than the operations per the current TS (where the fuel transfer canal gate would need to continually be removed and reinstalled throughout loading operations to maintain the current 16 feet minimum water coverage above fuel, per TS 4.1.2.a.).

Please provide an estimate of occupational doses, using applicable dose rates and anticipated time for each phase of the loading operations, expected to be accrued during: (a) the proposed operations per the license amendment request and proposed changes to TS (i.e., lower water levels in the FESW); and (b) the operations per the current TS (i.e., continued removal/reinstallation of fuel transfer canal gate throughout operations to maintain 16 feet minimum water coverage above fuel, per current TS 4.1.2.a).

Response to RAI 8

The required operations with and without the requested TS FESW level change have been reviewed and a dose estimated performed for both cases. The dose estimates are shown in the tables below.

Occupational Dose Estimate during Cask Loading Operations
With 16 Feet Minimum Water Coverage

Water Coverage	Task	Personnel	Duration (hours)	Dose Rate (mRem/hr)	Task Dose (mRem)	
16'	Assemble canal gate unbolting tool	3	0.50	1.50	2.250	
16'	Loosen 65 jacking bolts	3	4.00	1.50	18.000	
16'	Attach rigging to canal gate	3	0.25	1.50	1.125	
16'	Raise canal gate from guides	3	0.50	1.50	2.250	
16'	Move canal gate to storage	3	0.25	1.50	1.125	24.750

16’	Fuel moves to TSC	2	6.00	1.50	18.000	
16’	Fuel verification	2	6.00	1.50	18.000	
16’	Install DFC lids	2	2.00	1.00	4.000	
16’	Install TSC lid, remove rigging	3	1.00	1.00	3.000	43.000
16’	Attach rigging to canal gate	3	0.25	1.50	1.125	
16’	Remove canal gate from storage	3	0.25	1.50	1.125	
16’	Move canal gate to gate guides	3	0.25	1.50	1.125	
16’	Install canal gate	3	0.50	1.50	2.250	
16’	Tighten 65 jacking bolts	2	4.00	1.50	12.000	17.625

Dose per canal gate cycle (excluding fuel moves) (person-mRem)		42.375
Required number of canal gate cycles	7	
Total dose attributable to canal gate cycles (person-mRem)		296.625

Occupational Dose Estimate during Cask Loading Operations With 11 Feet, 6½ Inches Minimum Water Coverage

Water Coverage	Task	Personnel	Duration (hours)	Dose Rate (mRem/hr)	Task Dose (mRem)	
11’-6½”	Assemble canal gate unbolting tool	3	0.50	2.00	3.000	
11’-6½”	Loosen 65 jacking bolts	3	4.00	2.00	24.000	
11’-6½”	Attach rigging to canal gate	3	0.25	2.00	1.500	
11’-6½”	Raise canal gate from guides	3	0.50	2.00	3.000	
11’-6½”	Move canal gate to storage	3	0.25	2.00	1.500	33.000
16’	Fuel moves to TSC	2	6.00	1.50	18.000	
16’	Fuel verification	2	6.00	1.50	18.000	
16’	Install DFC lids	2	2.00	1.00	4.000	
16’	Install TSC lid, remove rigging	3	1.00	1.00	3.000	43.000
11’-6½”	Attach rigging to canal gate	3	0.25	2.00	1.500	
11’-6½”	Remove canal gate from storage	3	0.25	2.00	1.500	
11’-6½”	Move canal gate to gate guides	3	0.25	2.00	1.500	
11’-6½”	Install canal gate	3	0.50	2.00	3.000	
11’-6½”	Tighten 65 jacking bolts	2	4.00	2.00	16.000	23.500

Dose per canal gate cycle (excluding fuel moves) (person-mRem)		56.500
Required number of canal gate cycles	3	
Total dose attributable to canal gate cycles (person-mRem)		169.500

Summarizing:

Dose attributable to 7 fuel transfer canal gate cycles required with 16 feet minimum water coverage (person-mRem)	296.625
Dose attributable to 3 fuel transfer canal gate cycles required with 11 feet, 6½ inches minimum water coverage (person-mRem)	169.500
Occupational dose savings from reduced water coverage and canal gate cycles (person-mRem)	127.125

In the preceding tables, personnel dose estimates are shown for fuel loading operations with the current 16 feet minimum water coverage and 11 feet, 6½ inches minimum water coverage. Reduced water coverage permits a reduction in the number of fuel transfer canal gate cycles (removal and installation) thus realizing an estimated occupational dose savings of 127.125 person-mRem.

RAI 9

In the discussion of the impacts on accident analysis, the license amendment request notes that the level of water in the FESW is not a factor in this accident analysis since the accident analysis of record does not credit decontamination in the FESW water. The NRC staff reviewed the accident analysis as described in the December 2008 Decommissioning Plan (DP) and requests the following information regarding the analysis of record.

The spent fuel accident analyses described in the DP only consider the release of krypton-85 for which water decontamination is not a consideration. The analyses do not consider the release of iodine-129 (^{129}I) for which water decontamination would be a factor.

a. Please provide additional information describing the basis for not considering the release of ^{129}I in the spent fuel accident analyses described in the DP. Please provide additional information describing the impact of the reduced water level if the release of ^{129}I is considered in the dose consequence analysis.

The NRC staff notes that the dose consequences of the spent fuel accidents described in the DP are compared to the 25 rem whole body dose from 10 CFR Part 100. The dose consequences from fuel handling accidents are expected to be well within the limits of Part 100. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (SRP) Section 15.7.4, "Radiological Consequences of Fuel Handling Accidents," defines "well within" as 25 percent or less of the 10 CFR Part 100 exposure guideline values (i.e., 75 rem for the thyroid and 6 rem for the whole-body doses).

b. Please provide additional information to justify the use of an acceptance criteria value of 25 rem whole body rather than the SRP value of 6 rem whole body.

Response to RAI 9a

The following discussion refers to Reference 1.

The principal fission gas remaining for any potential fuel damage accident at LACBWR is krypton-85 because the reactor was permanently shut down over 20 years ago. Other krypton and xenon radioisotopes that were produced during plant operation have decayed to insignificant levels. Halogen radionuclides, such as radiobromines and radioiodines, including I-131, have also decayed to insignificant levels. Iodine-129, which has a half life of approximately 15.9 Million years, still remains in the LACBWR spent fuel assemblies. We have computed the total I-129 inventory in the FESW to be less than 0.4 Curie.

In the radiological analysis described in Reference 1, an iodine decontamination factor of 100 was assumed based on the level of water required to be maintained over the spent fuel assemblies (currently 16 feet). With this decontamination factor, a release rate of 1.82×10^{-7} Ci/sec I-129 was postulated to occur over a 2-hour duration and found to have an immeasurable thyroid and whole body dose to onsite personnel and to members of the public.

With the proposed reduction of 16 feet of water coverage over spent fuel assemblies to 11 feet 6½ inches (less than a 50% reduction in coverage), it can be conservatively postulated that the iodine decontamination factor is halved, from 100 to 50. This would result in a doubling of the iodine release rate previously considered, or a release rate of 3.64×10^{-7} Ci/sec. Even with the doubled release rate, the dose to onsite personnel and members of the public remains insignificant.

Response to RAI 9b

The LACBWR plant was first licensed in the late 1960s, well before the Standard Review Plan (SRP) was issued. The SRP was first issued in 1975 as NUREG-75/087 and subsequently re-issued as NUREG-0800 in 1981. Because LACBWR was licensed before the SRP existed, it was required to meet the dose limits in 10 CFR 100 for accident events, rather than the fractions of those limits published in the current revision SRP 15.7.4. The full Part 100 dose limits are documented as the acceptance criteria for radiological accidents in the LACBWR D-Plan, Sections 9.2 and 9.3. The NRC also accepted the full Part 100 dose limits in their Safety Evaluation Report for LACBWR operating license amendment 18 (Reference 2, Sections 3.6.1 and 3.6.2).

A comparison of the calculated accident event doses to the "well-within" SRP criteria was also performed in an effort to provide a comprehensive response to this RAI. The estimated doses for the radiological accidents described in Section 9.2 and 9.3 of the D-Plan, based on the source terms as of October 2008, are all less than 1 Rem at the Exclusion Area Boundary, with the maximum calculated dose being 0.8 Rem. The highest overall calculated dose is at the emergency planning zone boundary for a ground-level release after a shipping cask or heavy load drop into the FESW and is 3.8 Rem. This is less than the "well-within" whole-body dose limit in the SRP of 25 percent of the Part 100 limit, or 6 Rem.

REFERENCES

1. DPC-LACBWR Technical Report LAC-TR-134, "Determination of the Potential Onsite Radiological Consequences from a Fuel Handling Accident at the La Crosse Nuclear Facility," dated April, 1988.

2. NRC Safety Evaluation Report supporting LACBWR operating license amendment 18, dated July 13, 1979, and revised February 4, 1980.

3. LACBWR Decommissioning Plan, revised December 2009.

4. LACBWR Special Test Procedure, STP-58-01, "Perform Radiation Survey in the FESW," Issue 2. (Attached)

5. Sargent & Lundy Calculation No. 2010-04016, "Fuel Element Storage Well Pump Vortexing," Rev. 0, dated May 17, 2010. (Attached)

6. Weight Loss Data for LACBWR Fuel Storage Rack Poison Surveillance Program, revised September 4, 2008. (Attached)

DOCKET NO. 50-409

APPENDIX A

LICENSE NO. DPR-45

TECHNICAL SPECIFICATIONS FOR LA CROSSE BOILING WATER REACTOR

TABLE OF CONTENTS

1. DEFINITIONS

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The following terms are defined so that uniform interpretation of these specifications may be achieved. When these terms appear in capitalized type, the following definitions apply in these Technical Specifications.

ACTION

ACTION shall be that part of a specification which prescribes remedial measures required under designated conditions.

CHANNEL CALIBRATION

A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel outputs such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel including the sensor and the alarm and/or trip functions. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is calibrated.

FUEL HANDLING

FUEL HANDLING shall be the movement of individual spent fuel assemblies within the Reactor Building. Suspension of FUEL HANDLING shall not preclude completion of movement of a spent fuel assembly to a safe, conservative position. FUEL HANDLING, for the purposes of these Technical Specifications, does not include the movement of an NRC-certified spent fuel storage canister, transfer cask, or storage cask containing spent fuel in accordance with the dry cask storage system's 10 CFR 72 Certificate of Compliance.

OPERABLE-OPERABILITY

A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s) and when all necessary attendant instrumentation, controls, a normal or an alternate electrical power source, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

2. DESIGN FEATURES

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2.1 SITE

EXCLUSION AREA

2.1.1 The exclusion area shall be as described in the Off-Site Dose Calculation Manual.

2.2 FUEL STORAGE WHILE IN THE FUEL ELEMENT STORAGE WELL

CRITICALITY

2.2.1 The spent fuel storage racks are designed with a nominal 7.0 inch center- to-center distance between fuel assemblies in each individual rack assembly, with a boron impregnated poison plate between adjacent storage locations to ensure K_{eff} of $\leq$ 0.95 when flooded with unborated water.

FUEL RESTRICTIONS

2.2.2 Fuel stored in the storage well is restricted to fuel with stainless steel cladding which has a U-235 loading of $\leq$ 22.6 grams per axial centimeter of fuel assembly.

DRAINAGE

2.2.3 The Fuel Element Storage Well is designed and shall be maintained to prevent an inadvertent draining of the well below elevation of 679 feet MSL while spent fuel assemblies are in the Fuel Element Storage Well.

CAPACITY

2.2.4 The Fuel Element Storage Well was designed for a storage capacity of no more than 440 fuel assemblies. The maximum number of fuel assemblies stored in the Fuel Element Storage Well is limited to 333 spent fuel assemblies.

3. APPLICABILITY

LIMITING CONDITION FOR OPERATION

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3.1 Limiting Conditions for Operation and ACTION requirements shall be applicable during the specified applicable condition for each specification.

3.2 Adherence to the requirements of the Limiting Condition for Operation and/or associated ACTION within the specified time interval shall constitute compliance with the specification. In the event the Limiting Condition for Operation is restored prior to expiration of the specified time interval, completion of the ACTION statement is not required.

3.3 Entry into specified applicability state shall not be made unless the conditions of the Limiting Condition for Operation are met without reliance on provisions contained in the ACTION statements unless otherwise excepted.

SURVEILLANCE REQUIREMENTS

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3.4 Surveillance Requirements shall be applicable during the specified applicable conditions for individual Limiting Conditions for Operation unless otherwise stated in an individual Surveillance Requirement.

3.5 Each Surveillance Requirement shall be performed within the specified surveillance interval with a maximum allowable extension not to exceed 25 percent of the specified surveillance interval.

3.6 Performance of a Surveillance Requirement within the specified time interval shall constitute compliance with OPERABILITY requirements for a Limiting Condition for Operation and associated ACTION statements unless otherwise required by the specification. Surveillance requirements do not have to be performed on inoperable equipment or on equipment not required to be OPERABLE.

3.7 Entry into a specified applicable condition shall not be made unless the Surveillance Requirement(s) associated with the Limiting Condition for Operation have been performed within the stated surveillance interval or as otherwise specified.

4/5. PERFORMANCE REQUIREMENTS

4.1 FUEL STORAGE AND HANDLING

4.1.1 GENERAL FUEL STORAGE AND HANDLING REQUIREMENTS

4.1.1.1 Spent fuel assemblies shall be stored underwater in spent fuel storage racks that are positioned on the bottom of the Fuel Element Storage Well or in an approved dry spent fuel storage cask.

4.1.1.2 During the handling of spent fuel assemblies that have been operated at power levels greater than 1 Mwt, the depth of water in the Fuel Element Storage Well and the contiguous cask pool shall be at least 2 feet above the active fuel, and only one spent fuel assembly will be moved at a time.

4.1.1.3 No object heavier than 25 tons shall be handled over spent fuel assemblies located in the Fuel Element Storage Well or cask pool. Lifting and movement of a fuel-loaded storage canister and transfer cask shall be performed using the single-failure-proof cask handling crane lifting system meeting the guidance in NUREG-0612, Section 5.1.6. Lifting and movement of objects over spent fuel assemblies located in the Fuel Element Storage Well or cask pool shall be performed in accordance with the LACBWR NUREG-0612 commitments and the dedicated project heavy load control plan.

4.1.2 FUEL ELEMENT STORAGE WELL AND CASK POOL

LIMITING CONDITION FOR OPERATION

Note

This LCO does not apply to the cask pool if the spent fuel storage canister lid is in place in the canister or if there are no spent fuel assemblies in the cask pool.

The Fuel Element Storage Well (FESW) and cask pool shall meet the following requirements:

a. The Fuel Element Storage Well and cask pool water level shall be at least 11 feet, 6½ inches above any spent fuel assembly stored in the spent fuel storage racks or in spent fuel storage canister in the cask pool, and

b. Water in the storage well and cask pool shall be maintained at a temperature $\leq 150^{\circ}F$.

APPLICABILITY: While spent fuel assemblies are in the FESW or the cask pool.

ACTIONS

a. With water level less than required by the LCO, take immediate action to restore water level and suspend all operations involving FUEL HANDLING.

b. With water temperature in the storage well or cask pool above $150^{\circ}F$, take actions to reduce water temperature to $\leq 150^{\circ}F$ within 24 hours and suspend all operations involving FUEL HANDLING.

SURVEILLANCE REQUIREMENTS

Note

SR 5.1.2.1 and 5.1.2.2 do not apply to the cask pool if the spent fuel storage canister lid is in place in the canister or if there are no spent fuel assemblies in the cask pool.

5.1.2.1 The Fuel Element Storage Well and cask pool water level and temperature shall be verified at least once per 12 hours.

5.1.2.2 The Fuel Element Storage Well and cask pool water level indication channel shall be calibrated (CHANNEL CALIBRATION) at least once per 18 months.

6. ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The Plant Manager shall be responsible for overall facility operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 A Control Room Operator shall be responsible for the Control Room command function.

6.2 ORGANIZATION

6.2.1 FACILITY STAFF

6.2.1.1 The facility organization shall be as follows:

a. Each on-duty shift shall be composed of at least one Certified Fuel Handler and one qualified Control Room Operator when fuel is stored in the Fuel Element Storage Well.*

b. A qualified Control Room Operator shall be within visual and/or audio distance of the Control Room annunciators when fuel is in the Fuel Element Storage Well.

c. All FUEL HANDLING shall be directly supervised by a Certified Fuel Handler.

d. An individual qualified in radiation protection procedures shall be on site when there is fuel on site or there is a potential for release of radioactive materials. At least one additional Operator and one Health Physics Technician shall be on site when spent fuel or a spent fuel shipping cask is being handled or when any evolutions are being conducted in or above the Fuel Element Storage Well.

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* Shift crew composition may be one less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements. This provision does not permit any shift crew position to be unfilled upon shift change due to an oncoming shift crew member being late or absent.

6.2.1.2 OVERTIME POLICY

The working hours of Operators, Certified Fuel Handlers, Mechanical Maintenance and Instrument & Electrical Technicians when performing duties which may affect nuclear safety, and Health Physics Technicians, when performing radiation protection duties which may affect the safety of the public, shall be limited.

In the event overtime must be used, the following restrictions shall be followed:

(1) The specified personnel shall not be permitted to work more than 16 hours straight, excluding shift turnover time.

(2) The specified personnel shall not be permitted to work more than 16 hours in any 24-hour period, more than 24 hours in any 48-hour period, nor more than 72 hours in any 7-day period.

(3) A break of at least 8 hours shall be allowed following overtime before the next scheduled shift for the specified personnel, if the above limits are exceeded.

In the event overtime must be used in excess of the above restrictions, the Plant Manager or his designate, must authorize the deviation and the cause must be documented.

6.3 FACILITY STAFF QUALIFICATIONS

6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions except for the Health Physics Supervisor who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975.

6.4 PROGRAM REQUIREMENTS

6.4.1 In addition to the programs required by regulations, the programs specified in Section 6.4.2 shall be established, implemented and maintained.

6.4.2 PROGRAMS

6.4.2.1 PROCESS CONTROL PROGRAM (PCP)

a. The PCP shall be maintained on site and will be available for NRC review.

b. Licensee-initiated changes to the PCP shall be submitted to the Commission in the annual Radioactive Effluent Release Report for the period in which the change(s) was made. This submittal shall contain:

- Information to support the rationale for the change;
- A determination that the change did not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes; and
- Documentation of the fact that the change has been reviewed and found acceptable by the ORC.

6.4.2.2 OFFSITE DOSE CALCULATION MANUAL (ODCM)

The ODCM shall be maintained by the licensee. Changes to the ODCM will be outlined in the annual Radioactive Effluent Release Report per Specification 6.5.1.1.d.

This submittal shall contain:

(1) Detailed information to support the rationale for the change. Information submitted should consist of a package of those pages of the ODCM to be changed with each page numbered and provided with an approval and date box, together with appropriate analyses or evaluations justifying the change(s) and

(2) A determination that the change will not reduce the accuracy or reliability of dose calculations or setpoint determinations.

6.4.2.3 RADIOACTIVE EFFLUENT CONTROLS PROGRAM

A program shall be provided conforming with 10 CFR 50.36a for control of radio- active effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program (1) shall be contained in the ODCM, (2) shall be implemented by operating procedures, and (3) shall include remedial action to be taken whenever the program limits are exceeded. The program shall include the following elements:

(1) Limitations on the operability of radioactive liquid and gaseous monitoring instrumentation, including surveillance tests and setpoint determination in accordance with the methodology in the ODCM.

(2) Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas conforming to 10 CFR, Part 20, Appendix B, Table 2, Column 2.

(3) Monitoring, sampling and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20 and with the methodology and parameters in the ODCM.

(4) Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released to unrestricted areas conforming to Appendix I to 10 CFR, Part 50.

(5) Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every year.

(6) Limitations on the annual or quarterly air doses resulting from noble gases released in gaseous effluents to areas beyond the site boundary conforming to Appendix I to 10 CFR, Part 50.

(7) Limitations on the annual and quarterly doses to a member of the public from tritium and all radionuclides in particulate form with half-lives greater than eight days in gaseous effluents released to areas beyond the site boundary conforming to Appendix I to 10 CFR, Part 50.

(8) Limitations on the annual dose or dose commitment to any member of the public due to release of radioactivity and to radiation from uranium fuel cycle sources conforming to 40 CFR, Part 190.

6.4.2.4 RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

A program shall be provided to monitor radiation and radionuclides in the environs of the plant. The program shall provide representative measurements of radioactivity in the highest potential exposure pathways. The program shall (1) be contained in the ODCM, (2) conform to the guidance of Appendix I to 10 CFR, Part 50 and (3) include the following:

(1) Monitoring, sampling, analysis and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters in the ODCM.

(2) Participation in an Interlaboratory Comparison Program to ensure that independent checks on the precision and accuracy of the measurements of radioactive material in the environmental sample matrices are performed as part of the Quality Assurance Program for environmental monitoring.

6.5 REPORTING REQUIREMENTS

6.5.1 ROUTINE REPORTS

In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Regional Administrator of the Regional Office of the NRC unless otherwise noted.

6.5.1.1 Reports required on an annual basis shall be submitted by March 1 of each year and shall include:

a. A tabulation on an annual basis of the number of station, utility and other personnel, including contractors, receiving exposures greater than 100 mRem/yr and their associated man rem exposure according to work and job functions, e.g., plant operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and fuel handling. The dose assignment to various duty functions may be estimates based on pocket dosimeter, TLD, or film badge measurements. Small exposures totaling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific major work functions. This tabulation is per the requirements of Regulatory Guide 1.16, Revision 4, August 1975.

b. A report containing a brief description of any changes, testing and experiments conducted under the criteria of 10 CFR 50.59, including a summary of the safety evaluations of them.

c. An Annual Radiological Environmental Monitoring Report which shall include summarized and tabulated results, including interpretations and analysis of data trends, of environmental samples taken during the previous calendar year. In the event that some results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

The report shall also include the following: a summary description of the Radiological Environmental Monitoring Program; a map of all sampling locations keyed to a table giving distances and directions from the plant, the results of the Interlaboratory Comparison Program, and a discussion of all analyses in which the LLD was not achievable.

d. Radioactive Effluent Release Report

Paragraph (a)(2) of Part 50.36a, "Technical Specifications on Effluents from Nuclear Power Reactors," of 10 CFR Part 50 requires that a report be made to the Commission annually. The report shall specify the quantity of each of the principal radionuclides released to unrestricted areas by liquid and gaseous effluents during the previous year. With the exception of the collection of hourly meteorological data, the information submitted shall be in accordance with Appendix B of Regulatory Guide 1.21 (Revision 1) dated June 1974 with data summarized on at least a quarterly basis.

This same report shall include an assessment, performed in accordance with the Offsite Dose Calculation Manual (ODCM), of radiation doses to members of the public from radioactive liquid and gaseous effluents released beyond the effluent release boundary. This report shall contain any changes made to the ODCM during the previous twelve months.

6.6 <u>HIGH RADIATION AREA</u>

6.6.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.1601(a) of 10 CFR 20, each high radiation area in which the intensity of radiation, at 30 cm from the radiation source or surface that the radiation penetrates, is greater than 100 mrem/hr but less than 1000 mrem/hr shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Special Work Permit (SWP).* Any individual or group of individuals permitted to enter such areas shall be provided with one or more of the following:

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

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* Health Physics personnel or personnel escorted by Health Physics personnel shall be exempt from the SWP issuance requirement during the performance of their assigned radiation protection duties, provided they are following plant radiation protection procedures for entry into high radiation areas.

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel have been made knowledgeable of them.

c. A health physics qualified individual (i.e., qualified in radiation protection procedures) with a radiation dose rate monitoring device and who is responsible for providing positive exposure control over the activities within the area and who will perform periodic radiation surveillance at the frequency which will be established by the Health and Safety Supervisor or applicable SWP.

6.6.2 For each area with radiation levels greater than 1000 mrem/hr, at 30 cm (but less than 500 Rad/hr at 1 meter) from radiation source, or from any surface penetrated by the radiation, the control of Specification 6.6.1 shall be implemented and also:

(1) Each entrance or access point to the area shall be maintained locked except during periods when access to the area is required. Positive control over each individual entry shall be by:

a. Maintaining the locked door keys under administrative control of the Certified Fuel Handler on duty or the Health and Safety Supervisor.

b. An approved SWP that specifies the dose rates in the immediate work areas and the maximum allowable stay time for individuals in that area.

PERFORM RADIATION SURVEY IN THE FESW

NON-CONTROLLED COPY

RECORD AND CONTROL OF INITIAL ISSUE, REVISIONS & PERIODIC REVIEWS										
ISSUE	PREPARED BY OR REVIEWED BY		OPERATIONS SUPERVISOR REVIEW		H & S ** REVIEW		QA REVIEW		APPROVED & ISSUED* EFFECTIVE DATE OF ISSUE	
	SIGNATURE	DATE	SIGNATURE	DATE	SIGNATURE	DATE	SIGNATURE	DATE		
0	[signature]	6/8/09	[signature]	6/8/09	Seymour Rafferty	6-12-09	[signature]	6/15/09	[signature]	6/22/09
1	[signature]	4/1/10	R Cota	4/6/10	Seymour Rafferty	4-6-10	[signature]	4/6/10	[signature]	4/6/10
2	[signature]	4/19/10	Daniel [signature]	4-19-10	Seymour Rafferty	4-19-10	[signature]	4/19/10	[signature] for M Berard	4/20/10

* LACBWR PLANT MANAGER
** H&S SUPERVISOR OR RAD. PROT. ENGINEER

LACBWR

SPECIAL TEST PROCEDURE

PERFORM RADIATION SURVEY IN THE FESW

Issue Notice No. 2 Dated APR 20 2010

INSTRUCTIONS

	Description of and Reason for Change
REMOVE AND INSERT ALL PAGES	Modify procedure to obtain survey data various levels above the fuel racks.
Changes to this procedure do require 50.59 screening.	

This issue shall not become effective unless accompanied by a new cover sheet, properly signed off in the appropriate review/approval columns.

LACBWR

SPECIAL TEST PROCEDURE

PERFORM RADIATION SURVEY IN THE FESW

1.0 PURPOSE

This procedure provides guidance for performing an underwater radiation survey in the FESW. This survey will be done at various levels above the fuel racks, as directed, to determine radiation dose rates from the fuel at this level.

2.0 PRE-REQUISITES

2.1 SWP available.

2.2 MR completed.

2.3 FESW pool covers removed.

2.4 Contaminated work area established around pool.

2.5 Pre-job briefing performed.

2.6 Perform pre-operations checks on the underwater probe and instrument.

3.0 REFERENCES

3.1 ACP-18.1, "Collection, Storage and Maintenance of LACBWR Quality Assurance Records"

3.2 HSP-02.6, "Radiation Surveys"

4.0 PROCEDURE

4.1 All pre-requisites have been completed.

4.2 HP will lower the underwater probe to the level determined as a second person should verify the probe is at the desired depth.

4.3 Move the detector to the pre-determined survey location.

4.4 Allow meter reading to stabilize.

4.5 Record reading on Attachment 1.

4.6 Repeat steps 4.3 through 4.5 until all survey points have been taken.

4.7. Wipe down and restore the underwater probe and survey instrument.

4.8 Take completed survey form to the Health & Safety/Maintenance Supervisor.

5.0 RECORDS

All records generated using this procedure will be maintained in accordance with ACP-18.1.

6.0 ATTACHMENTS

6.1 Attachment 1 – FESW Underwater Survey

ATTACHMENT 1
Page 1 of 1

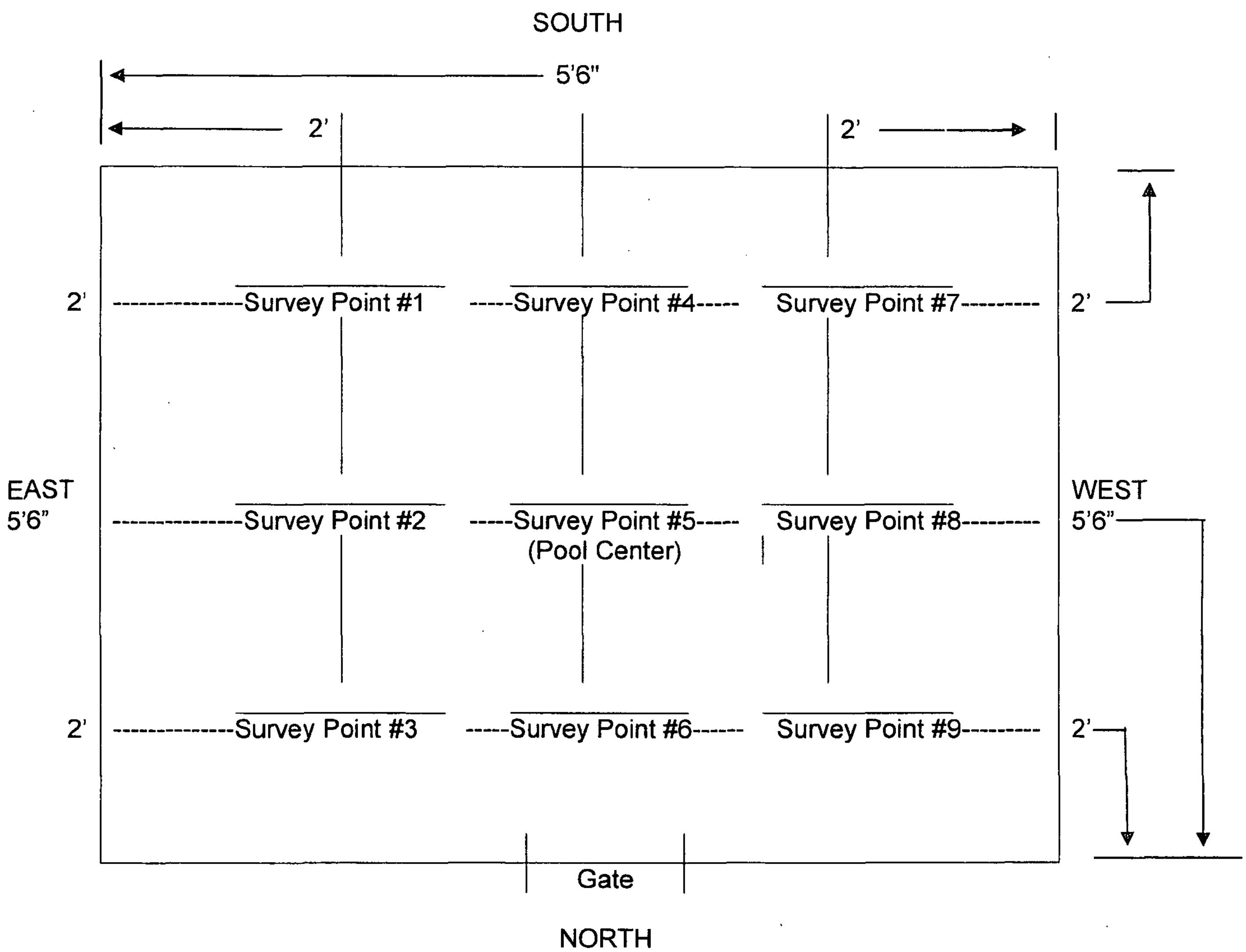
FESW Underwater Survey

Date ____________________ Performed by ______________________
Print

____
Signature

Survey Instrument Serial#, Make & Model# ______________________________

Depth from 701 feet minus ________ feet = ______ feet Instrument Check Point ______ R/hr
Before ______ R/hr
After ______ R/hr



Top of Fuel Racks Elevation 678'

Data verified by: ______________________
(print name)

____
(signature)

CONTROLLED
FIELD COPY
4/26/10

COPY

ATTACHMENT 1
Page 1 of 1

FESW Underwater Survey

Date 4/27/10

Performed by PENNEBECKER
Print

B. E. Pennebecker
Signature

Survey Instrument Serial#, Make & Model# SERIAL 021020 THERMO SCIENTIFIC FH 406-X

Depth from 701 feet minus 13 feet = 688 feet (10' above fuel)

Instrument Check Point 13.0 R/hr
Before 13.0 R/hr
After 13.2 R/hr

SOUTH

5'6"

	2'		2'
2'	<10 mRem/HR Survey Point #1	<10 mRem/HR Survey Point #4	<10 mRem/HR Survey Point #7
EAST 5'6"	<10 mRem/HR Survey Point #2	<10 mRem/HR Survey Point #5 (Pool Center)	<10 mRem/HR Survey Point #8
2'	<10 mRem/HR Survey Point #3	<10 mRem/HR Survey Point #6	<10 mRem/HR Survey Point #9

2'
WEST 5'6"
2'

Gate

NORTH

Top of Fuel Racks Elevation 678'

Data verified by: Philip Berg / BRIAN WHITE
(print name)

Philip W Berg / B White
(signature)

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FIELD COPY
4/26/10

STP-58-01
Issue 2

ATTACHMENT 1
Page 1 of 1

COPY

FESW Underwater Survey

Date 4/27/10

Performed by PENNEBECKER
Print

R.E. Pennebecker
Signature

Survey Instrument Serial#, Make & Model# SERIAL 021020 THERMO SCIENTIFIC FH 40 G-X

Depth from 701 feet minus 15 feet = 686 feet (8' above fuel)

Instrument Check Point	13.0 R/hr
Before	13.0 R/hr
After	13.2 R/hr

SOUTH

5'6"

2' / 2'

2'	<10 mRem/HR Survey Point #1	<10 mRem/HR Survey Point #4	<10 mRem/HR Survey Point #7	2'
EAST 5'6"	<10 mRem/HR Survey Point #2	<10 mRem/HR Survey Point #5 (Pool Center)	<10 mRem/HR Survey Point #8	WEST 5'6"
2'	<10 mRem/HR Survey Point #3	<10 mRem/HR Survey Point #6	<10 mRem/HR Survey Point #9	2'

Gate

NORTH

Top of Fuel Racks Elevation 678'

Data verified by: Philip Borg / BEAN WACH
(print name)
Philip W Borg / [signature]
(signature)

CONTROLLED FIELD COPY
4/26/10

STP-58-01
Issue 2

ATTACHMENT 1
Page 1 of 1

COPY

FESW Underwater Survey

Date 4/27/10

Performed by PENNEBECKER
Print

R.E. Pennebecker
Signature

Survey Instrument Serial#, Make & Model# SERIAL 021020 THERMO SCIENTIFIC FH 40G-X

Depth from 701 feet minus 17 feet = 684 feet
(6' above fuel)

Instrument Check Point 13.0 R/hr
Before 13.0 R/hr
After 13.2 R/hr

SOUTH

5'6"

2' | 2'

EAST 5'6"				WEST 5'6"
2'	<10 mRem/HR Survey Point #1	<10 mRem/HR Survey Point #4	<10 mRem/HR Survey Point #7	2'
	11.1 mRem/HR Survey Point #2	<10 mRem/HR Survey Point #5 (Pool Center)	<10 mRem/HR Survey Point #8	
2'	<10 mRem/HR Survey Point #3	<10 mRem/HR Survey Point #6	<10 mRem/HR Survey Point #9	2'

Gate

NORTH

Top of Fuel Racks Elevation 678'

Data verified by: Philip Berg / Brian White
(print name)

Philip W Berg / B White
(signature)

CONTROLLED FIELD COPY
4/26/10

ATTACHMENT 1
Page 1 of 1

COPY

FESW Underwater Survey

Date 4/27/10 Performed by PENNEBECKER
Print

R. E. Pennebecker
Signature

Survey Instrument Serial#, Make & Model# SERIAL 021020 THERMO SCIENTIFIC FH-40G-X

Depth from 701 feet minus 19 feet = 682 feet (4' above fuel)

Instrument Check Point 13.0 R/hr
Before 13.0 R/hr
After 13.2 R/hr

SOUTH

5'6"

2' 2'

2'	<10 mRem/HR Survey Point #1	<10 mRem/HR Survey Point #4	11.9 mRem/HR Survey Point #7	2'
EAST 5'6"	14.3 mRem/HR Survey Point #2	<10 mRem/HR Survey Point #5 (Pool Center)	18.4 mRem/HR Survey Point #8	WEST 5'6"
2'	<10 mRem/HR Survey Point #3	17.6 mRem/HR Survey Point #6	22.3 mRem/HR Survey Point #9	2'

Gate

NORTH

Top of Fuel Racks Elevation 678'

Data verified by: Philip Berg / BRIAN WAITE
(print name)
Philip W Berg / B Waite
(signature)

ISSUE SUMMARY
Form SOP-0402-07, Revision 8

DESIGN CONTROL SUMMARY

CLIENT: Daiyland Power Cooperative — UNIT NO.: 1 — PAGE NO.: 1

PROJECT NAME: **La Crosse Boiling Water Reactor**

PROJECT NO.: 08785-080

CALC. NO.: 2010-04016

S&L NUCLEAR QA PROGRAM APPLICABLE ☒ YES ☐ NO

TITLE: Fuel Element Storage Well Pump Vortexing - Important to Safety B

EQUIPMENT NO.:

IDENTIFICATION OF PAGES ADDED/REVISED/SUPERSEDED/VOIDED & REVIEW METHOD

Prepared Pages 1-5, Attachments A, B, and C

INPUTS/ ASSUMPTIONS
☒ VERIFIED
☐ UNVERIFIED

REVIEW METHOD:	Detailed	REV.:	0
STATUS:	☒ APPROVED ☐ SUPERSEDED BY CALCULATION NO. ☐ VOID	DATE FOR REV.:	5/17/10
PREPARER:	Craig Thompson [signature]	DATE:	5/17/10
REVIEWER:	Eric DeCristofaro [signature]	DATE:	5/17/10
APPROVER:	Robert Peterson Robert J Peterson	DATE:	5-17-10

IDENTIFICATION OF PAGES ADDED/REVISED/SUPERSEDED/VOIDED & REVIEW METHOD

INPUTS/ ASSUMPTIONS
☐ VERIFIED
☐ UNVERIFIED

REVIEW METHOD:		REV.:	
STATUS:	☐ APPROVED ☐ SUPERSEDED BY CALCULATION NO. ☐ VOID	DATE FOR REV.:	
PREPARER:		DATE:	
REVIEWER:		DATE:	
APPROVER:		DATE:	

IDENTIFICATION OF PAGES ADDED/REVISED/SUPERSEDED/VOIDED & REVIEW METHOD

INPUTS/ ASSUMPTIONS
☐ VERIFIED
☐ UNVERIFIED

REVIEW METHOD:		REV.:	
STATUS:	☐ APPROVED ☐ SUPERSEDED BY CALCULATION NO. ☐ VOID	DATE FOR REV.:	
PREPARER:		DATE:	
REVIEWER:		DATE:	
APPROVER:		DATE:	

NOTE: PRINT AND SIGN IN THE SIGNATURE AREAS

1. Purpose and Scope

This calculation determines the Fuel Element Storage Well (FESW) pump intake requirements for no surface vortices to occur. The formation of strong free-surface air core vortices, commonly referred to as "vortexing", should not occur if the intake piping meets the parameters required by this calculation.

This calculation provides a curve determining the minimum centerline submergence of the intake pipe for any given flow rate. This will provide future flexibility for changes to water level and flow rate.

2. References

2.1. DT-SL-10-17, Document Transmittal from Dairyland Power Cooperative, 05/13/2010.
   - 2.1.1. Substitution Request for Piping System for Reactor Plant, Rev. 0, 06/16/1997.
   - 2.1.2. LACBWR Fuel Storage Well Flow Diagram, Rev. 23, 05/18/2007.
   - 2.1.3. OP Manual, Volume II, Section 11 "Reactor Process Systems", 12/21/2006.

2.2. ANSI/HI 9.8-1998, "American National Standard for Pump Intake Design," American National Standards Institute, Inc., November 1998.

2.3. Crane Technical Paper No. 410, "Flow of Fluids Through Valves, Fittings, and Pipe," Crane Co., 1988.

3. Inputs

3.1. The pump intake piping is 6 inch nominal pipe size (Ref. 2.1.2), schedule 40S (Ref. 2.1.1). The piping has an inner diameter of 6.065 inches or 0.505 feet (Ref. 2.3). This is the only design input used to generate the centerline submergence versus flow rate correlation curve.

3.2. The maximum FESW pump flow rate is 270 gpm (Ref. 2.1.3, Section 11.3.3).

4. Assumptions

None.

5. Methods

Minimum Required Submergence

Per the American National Standard for Pump Intake Design (Ref. 2.2), the minimum submergence is calculated based on the Froude number using Equation 5.1-1, shown below. If the left side of the equation is greater than the right side, then a vortex will not occur (Ref. 2.2, pg. 32).

$$\frac{S}{D} = 1.0 + 2.3 \times Fr_D \qquad \text{(Equation 5.1-1)}$$

Where:

- S - Submergence to the center of the pipe (ft)
- D - Pipe inner diameter (ft)
- Fr_D - Froude number $Fr_D = \frac{V}{\sqrt{g \times D}}$
- V - Flow velocity (ft/s)
- g - Acceleration due to gravity (32.2 ft/s^2)

For this particular application, Equation 5.1-1 can also be expressed as the following (Ref. 2.2, pg. 32):

$$S_{min} = D + 0.574\left(\frac{Q}{D^{1.5}}\right) \qquad \text{(Equation 5.1-2)}$$

Where:

- S_{min} - Minimum submergence to the center of the pipe (in)
- D - Pipe inner diameter (in)
- Q - Volume flow rate (gpm)

This equation is plotted against a range of flow rates to develop a correlation curve. The curve can be used to find the minimum submergence for any flow rate within the specified range.

6. Calculations and Results

6.1. Minimum Required Submergence

Equation 5.1-2 is the basis for the correlation curve of submergence and flow rate at a specific pipe inner diameter. Based on an intake pipe inner diameter of 6.065 inches (Input 3.1), the equation becomes:

$$S_{\min}(Q) = 6.065 + 0.574 * \left(\frac{Q}{6.065^{1.5}}\right) \qquad \text{(Equation 6.1-1)}$$

Where: S_{min} - Minimum submergence to the center of the pipe (in)
Q - Volume flow rate (gpm)

This relation is plotted over the range of flow rates determined in Section 6.2.

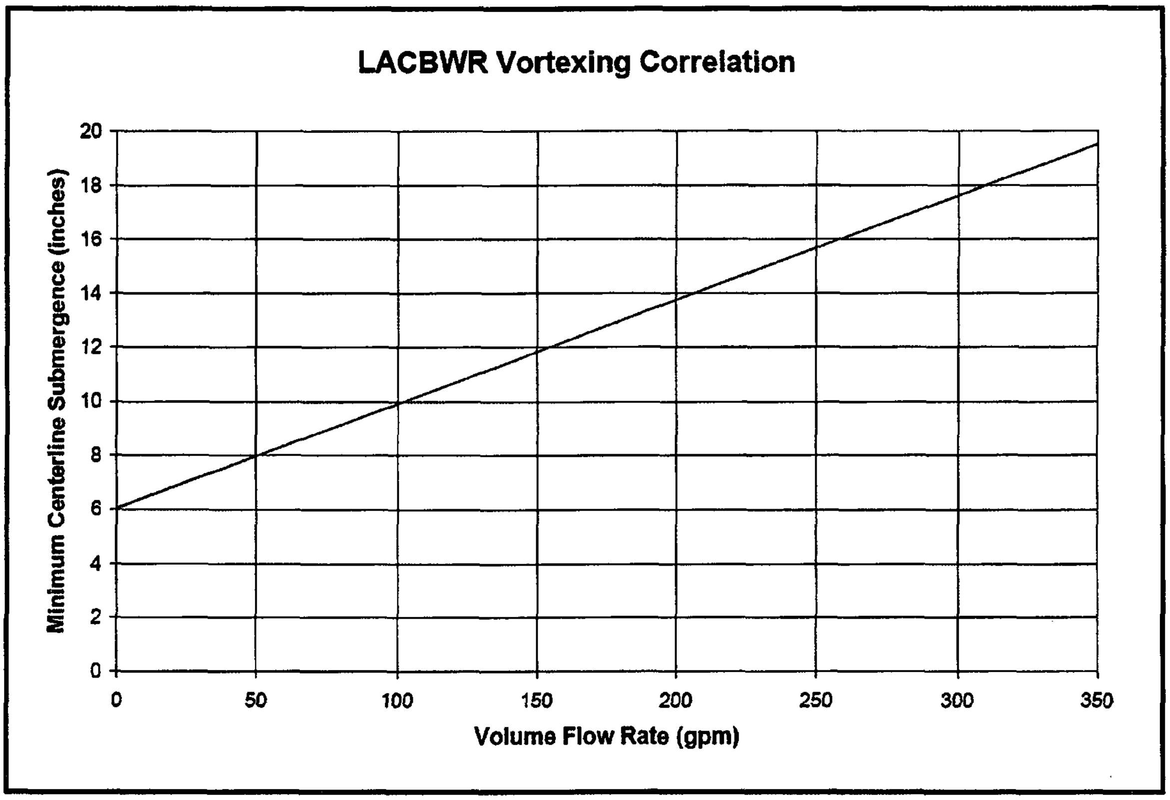
6.2. Range of Flow Rates

The maximum FESW pump flow rate is 270 gpm (Input 3.2). To provide flexibility for future FESW pump changes, Equation 6.1-1 is plotted over the range of 0-350 gpm and is shown in Attachment A.

This correlation curve can be used to determine the minimum submergence at a given flow rate for no vortexing to occur. The data for the above calculations and the correlation curve are shown in Attachments B and C.

7. Conclusions and Recommendations

For no vortexing to occur in the spent fuel pool circulation pump intake piping, the correlation curve in Attachment A should be observed. At a particular flow rate, the minimum submergence should be calculated using Equation 6.1-1.


LACBWR Vortexing Correlation
Minimum Centerline Submergence (inches)
0
2
4
6
8
10
12
14
16
18
20
0
50
100
150
200
250
300
350
Volume Flow Rate (gpm)

	A	B	C	D
2	Design Input	Parameter	Value	Units
3	3.1	Pipe Inner Diameter	6.065	inches
4				
5				
6	Flow Rate (gpm)	Centerline Submergence (inches)		
7	0.0	6.065		
8	10.0	6.449		
9	20.0	6.834		
10	30.0	7.218		
11	40.0	7.602		
12	50.0	7.986		
13	60.0	8.371		
14	70.0	8.755		
15	80.0	9.139		
16	90.0	9.524		
17	100.0	9.908		
18	110.0	10.292		
19	120.0	10.677		
20	130.0	11.061		
21	140.0	11.445		
22	150.0	11.829		
23	160.0	12.214		
24	170.0	12.598		
25	180.0	12.982		
26	190.0	13.367		
27	200.0	13.751		
28	210.0	14.135		
29	220.0	14.520		
30	230.0	14.904		
31	240.0	15.288		
32	250.0	15.672		
33	260.0	16.057		
34	270.0	16.441		
35	280.0	16.825		
36	290.0	17.210		
37	300.0	17.594		
38	310.0	17.978		
39	320.0	18.362		
40	330.0	18.747		
41	340.0	19.131		
42	350.0	19.515		

	A	B	C	D
2	Design Input	Parameter	Value	Units
3	3.1	Pipe Inner Diameter	6.065	inches
4				
5				
6	Flow Rate (gpm)	Centerline Submergence (inches)		
7	0	=$C$3+0.574*(A7/$C$3^1.5)		
8	10	=$C$3+0.574*(A8/$C$3^1.5)		
9	20	=$C$3+0.574*(A9/$C$3^1.5)		
10	30	=$C$3+0.574*(A10/$C$3^1.5)		
11	40	=$C$3+0.574*(A11/$C$3^1.5)		
12	50	=$C$3+0.574*(A12/$C$3^1.5)		
13	60	=$C$3+0.574*(A13/$C$3^1.5)		
14	70	=$C$3+0.574*(A14/$C$3^1.5)		
15	80	=$C$3+0.574*(A15/$C$3^1.5)		
16	90	=$C$3+0.574*(A16/$C$3^1.5)		
17	100	=$C$3+0.574*(A17/$C$3^1.5)		
18	110	=$C$3+0.574*(A18/$C$3^1.5)		
19	120	=$C$3+0.574*(A19/$C$3^1.5)		
20	130	=$C$3+0.574*(A20/$C$3^1.5)		
21	140	=$C$3+0.574*(A21/$C$3^1.5)		
22	150	=$C$3+0.574*(A22/$C$3^1.5)		
23	160	=$C$3+0.574*(A23/$C$3^1.5)		
24	170	=$C$3+0.574*(A24/$C$3^1.5)		
25	180	=$C$3+0.574*(A25/$C$3^1.5)		
26	190	=$C$3+0.574*(A26/$C$3^1.5)		
27	200	=$C$3+0.574*(A27/$C$3^1.5)		
28	210	=$C$3+0.574*(A28/$C$3^1.5)		
29	220	=$C$3+0.574*(A29/$C$3^1.5)		
30	230	=$C$3+0.574*(A30/$C$3^1.5)		
31	240	=$C$3+0.574*(A31/$C$3^1.5)		
32	250	=$C$3+0.574*(A32/$C$3^1.5)		
33	260	=$C$3+0.574*(A33/$C$3^1.5)		
34	270	=$C$3+0.574*(A34/$C$3^1.5)		
35	280	=$C$3+0.574*(A35/$C$3^1.5)		
36	290	=$C$3+0.574*(A36/$C$3^1.5)		
37	300	=$C$3+0.574*(A37/$C$3^1.5)		
38	310	=$C$3+0.574*(A38/$C$3^1.5)		
39	320	=$C$3+0.574*(A39/$C$3^1.5)		
40	330	=$C$3+0.574*(A40/$C$3^1.5)		
41	340	=$C$3+0.574*(A41/$C$3^1.5)		
42	350	=$C$3+0.574*(A42/$C$3^1.5)		

Weight Loss Data for LACBWR Fuel Storage Rack Poison Surveillance Program

(Initially installed October 19, 1980)

Sample & History		Date	Percent of Weight Change from Original Weight in 1980										
			July 22, 1982	February 6, 1984	May 23, 1985	June 12, 1986	June 30, 1987		December 17, 1990	February 16, 1996	March 14, 2001	April 11, 2005	September 4, 2008
Sample	Original Weight (grams)	Drying Method	Oven Dried ~250° F	Oven Dried ~250° F	Oven Dried ~200° F	Oven Dried ~200° F	Air Dried ~ 24 hours	Oven Dried ~2 hours @ ~200° F	Air Dried ~ 3 days	Air Dried ~24.5 hours	Air Dried ~26.5 hours	Air Dried ~68 hours	Air Dried ~26 hours
5A Inner	24.59000		-1.79	-2.44					-2.98				
5A Outer	24.6769		+5.89	-2.49					-3.02				
5B Inner	25.7748		+5.22		-2.97	-3.65				-4.37	*Removed March 6, 1997- Both samples sent to NETCO for testing. Samples have not been replaced on sample holder.		
5B Outer	24.1488		+5.54		-2.88	-2.76				-4.70			
5C Inner	24.7266		+4.94			-4.82					-4.07		
5C Outer	25.3507		+5.25			-2.16					-3.96		
5D Inner	25.7108		-1.08			-2.27	-2.37	-3.19				-10.7	
5D Outer	25.2461		+4.38			-3.56	-2.46	-3.32				-12.0	
6B Inner	24.8896			-2.77					-3.34				-5.17
6B Outer	25.6584			-2.68					-3.20				-5.53
6C Inner	24.4982				-2.66					-3.94			
6C Outer	24.2182				-2.83					-4.60			
6D Inner	25.4933					-3.47					-4.10		
6D Outer	25.4872					-2.84					-4.99		
6A Inner	25.9048						-2.43	-3.92				-11.7	
6A Outer	25.6718						-1.86	-2.83				-11.3	
7A Inner	25.0467								-3.21 *			-4.68 *	
7A Outer	25.6823								-2.79 *			-4.66 *	
7C Inner	24.8927								-2.87 *				
7C Outer	25.4753								-2.90 *				
7B Inner	25.2320											-4.83 *	
7B Outer	25.4866											-4.54 *	
7D Inner	24.9698												-4.78
7D Outer	24.3541												-5.30

*Air Dried ~45 hours. Weighed Dec. 19, 1990

*Air Dried ~26 hours, Weighed April 15, 2005