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STATE OF VERMONT
DEPARTMENT OF PUBLIC SERVICE

DOCKETED
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July 6, 2006 (3:54pm)

OFFICE OF SECRETARY
RULEMAKINGS AND
ADJUDICATIONS STAFF

July 6, 2006

Administrative Judge
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Entergy Nuclear Operations, Inc.
(Vermont Yankee Nuclear Power Station)
Docket No. 50-271-LR; ASLBP No. 06-849-03-LR

Dear Administrative Judges:

Please find enclosed for filing a Corrected Copy of the Vermont Department of Public Service Reply to Answers of Applicant and NRC Staff to Notice of Intention to Participate and Petition to Intervene. The original version was filed on June 30, 2006. That afternoon as we worked to file by close of business, the document we had been working on was somehow corrupted and could no longer be used. Symptoms included random freezing of the entire program, text disappearing unbidden, and material to be deleted being frozen into the document. The State's technical people are still trying to figure out what happened. That day we finally took an older version and manually inserted the changes as quickly as we could, and cut and pasted where we could.

After the filing was electronically mailed at about 6:40 PM on June 30, 2006, I took some time off. Upon my return it became obvious that unfortunately some mistakes were made in our haste to get the filing mailed that night. Accordingly, I am providing a "Corrected Copy Submitted on 7/6/06" correcting all the places the changes did not get

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made in the document filed that were in the copy that originally had been prepared for filing but was horribly corrupted.

On the Corrected Copy I have included margin indicators of where changes were made. On the electronic copy, the changes also show in red. Although the changes occur only in the introduction material and the argument for contention 1, I have included the entire Reply so that parties can discard their earlier copy and rely on this Corrected Copy. With the hard copies mailed tomorrow, will be the exhibits and declaration of William Sherman as well, although no changes were made to these documents. They will be included for convenience.

I am sorry for any inconvenience this entire document corruption caused. I am available to answer any questions you might have about this process.

Very truly yours,

Sarah Hofmann
Director for Public Advocacy
Vermont Department of Public Service

cc: See Attached Certificate of Service

UNITED STATES
NUCLEAR REGULATORY COMMISSION

In Re: Entergy Nuclear Vermont Yankee)
LLC and Entergy Nuclear)
Operations, Inc.)
(Vermont Yankee Nuclear Power Station))

Docket No. 50-271
ASLBP No. 06-849-03-LR

CORRECTED COPY SUBMITTED ON 7/6/06

VERMONT DEPARTMENT OF PUBLIC SERVICE REPLY TO ANSWERS OF APPLICANT AND NRC STAFF TO NOTICE OF INTENTION TO PARTICIPATE AND PETITION TO INTERVENE

INTRODUCTION

Although Applicant and the Staff[1] point to NRC regulations that establish some criteria on the extent to which a State has rights to participate once a hearing has been established (see 10 CFR §2.315 (c)), they do not and cannot dispute the fact that 42 U.S.C. §2021(l) guarantees

[1] Pursuant to 10 CFR §2.309(h)(1) an Answer to a Petition to Intervene may only be filed by the " applicant/licensee, the NRC staff, and any other party to a proceeding" (emphasis added). On its face, §2.309(h) limits answers to parties by following the listing of the Applicant and the NRC staff by the language "and any other party" (emphasis added). Thus, as used in §2.309(h), the Staff and Applicant, are deemed to be parties and the Staff's right to file an answer is dependent upon it having party status. In addition, the Commission has noted the Staff may not take a position on any issue in proceedings under Subpart L where the Staff is going to take action independent of the hearing process:

> In no event, however, should the staff's explanation set forth a position on, or otherwise assume an advocacy position with respect to the contested matter in the adjudication before the presiding officer. 69 Fed. Reg. 2182, 2228 (January 14, 2004) (Statement of Considerations).

By filing an Answer the Staff has elected to be a party. By taking a position on the merits of Contentions the Staff has determined it will not take action on the Proposed Amendment independent of the hearing process. We believe the Staff should be a party to this proceeding. Its decision to not act independently of the hearing process illustrates appropriate respect for the importance of the full exploration of the issues in a hearing, albeit a less full hearing than DPS believes is required.

every state that the NRC "shall afford reasonable opportunity for State representatives to offer evidence, interrogate witnesses, and advise the Commission as to the application" for any licensing amendment authorizing operation of a nuclear reactor whether or not a hearing is to be held. Nothing in §2.315 (c) purports to limit the rights created by the statute but merely specifies the procedures that apply to the participation of a state in a hearing that has been convened because other parties have admitted contentions. The legislative history of the statutory provision provides an equally clear and unlimited statement of the rights guaranteed:

> Subsection l. provides appropriate recognition of the interest of the States in activities which are continued under Commission authority. Thus, the Commission is required to give prompt notice to the States of the filing of license applications and to afford reasonable opportunity for State representatives to offer evidence, interrogate witnesses, and advise the Commission as to the application.

Senate Report No. 870,1959 U.S. Code Congressional and Administrative News, p 2883.

The rights provided by 10 CFR §2.315 (c) are not exercisable until after the Board has determined if there are any admissible contentions further underscoring the fact that §2.315(c) is not intended to fully implement all the rights provided by 42 U.S.C. § 2021(l). Otherwise, the Regulation would read out of existence the plain language of the statute which does not condition the right of the state to participate with respect to a licensing amendment on whether the NRC has identified an admissible contention. In fact, since the Statute explicitly grants the right to present evidence and cross-examination without the State ever having to take a position on the merits, a reading of the Regulation to prohibit the State from participation unless there were an admissible contention would, in a case where no other entity sought to challenge the proposed amendment, effectively deny the State a right explicitly guaranteed by the Statute. We believe

the Board should not read the Commission Regulations to be in direct conflict with a statutory mandate where the language of the statute is susceptible to an interpretation which does not conflict with the statute. In this case, §2.315 (c) should be read only to prescribe certain procedures to be followed in a situation where an entity other than the State has presented a contention which has been found to be admissible and then only as to the particular hearing convened for the purpose of resolving that contention. The far reaching reading of §2.315 (c) urged by Applicant and the Staff would put it in direct conflict with the AEA. The Board should not assume that the Commission has chosen to ignore the precise language of a statutory provision, particularly where the Commission does not indicate that in adopting §2.315(c) it was intending to foreclose any other application of the state rights guaranteed by 42 U.S.C. § 2101(l).

The AEA requires a hearing whenever a proposed amendment presents a "significant hazard consideration". 42 U.S.C. §2239. Because §2101(l) guarantees the State certain rights when there is a right to a hearing, it is important for the Board to determine whether the proposed amendment is one which requires a hearing in order to determine whether the rights guaranteed to the State under §2101(l) are applicable here. A critical factor used by the Courts in determining whether a significant hazard is present and whether a hearing is required is whether the proposed amendment provides the licensee with greater operating authority. *See In re Three Mile Island Alert, Inc.*, 771 F.2d 720 (3d Cir.1985) *TMI*), *cert. denied,* 475 U.S. 1082, 106 S.Ct. 1460, 89 L.Ed.2d 717 (1986)(where the Court distinguishes the case before it, where no hearing was required, because the licensee has given "no greater operating authority"(*id.* 771 F.2d at 729)) and *Kelley v. Selin*, 42 F.3d 1501 (6th Cir. 1995)(where the Court found no hearing was required because the actions proposed "do not grant Consumers the right to operate Palisades in

any greater capacity than the plant had previously been allowed to operate" (*id.* 42 F.3d at 1515)). In this case Applicant seeks a substantial alteration in its operating authority - the right to operate the plant for an additional 20 years - and thus the proposed amendment meets the Court recognized standard for when a hearing must be held.

The Commission has codified the factors to be evaluated in determining whether a significant hazard consideration is present in 10 CFR §50.92 (c):

> (c) The Commission may make a final determination, pursuant to the procedures in § 50.91, that a proposed amendment to an operating license for a facility licensed under § 50.21(b) or § 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:
>
> > (1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or
> > (2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or
> > (3) Involve a significant reduction in a margin of safety.

DPS has raised considerations involving at least two of these in its Contentions. First, the reduction of strength and modulus of concrete for primary containment concrete would create a significant potential hazard. Second, it may also involve a new or different kind of accident in that the NRC has never evaluated the consequences of the failure of primary containment concrete in the event of a major accident. The focus in determining whether a hearing must be held is not on whether the result of the hearing will be that no significant hazard was created but on whether a legitimate issue has been raised regarding the existence of a "significant hazard". In *San Luis Obispo Mothers for Peace v. NRC*, 799 F.2d 1268 (9th Cir. 1986) the Court held:

> The regulations thus appropriately require a hearing before the proposed license amendment becomes effective whenever the amendment *creates the possibility* of a new or different kind of accident. Petitioners have identified such an accident and they

should have been granted a prior hearing.

Id. 799 at 1270 (emphasis in original).

Therefore, DPS, as the designated representative of the State of Vermont, is entitled to the hearing rights guaranteed to it under 42 U.S.C. § 2101(l). Nothing in the statute or in NRC regulations limits the right of DPS to present evidence and cross-examine witnesses where, as here, a hearing must be held. Thus, even if the Board were to determine that no admissible contention has been presented and that pursuant to Part 2 no Subpart L or Subpart G hearing is to be held or even that an admissible contention has been presented but only a Subpart L hearing will be conducted, the rights guaranteed to DPS under the statute are not diminished. Although DPS could exercise its rights without identifying the topics on which it seeks to present such evidence and interrogate witnesses, DPS has chosen to define the specific subjects of interest to it by filing contentions and seeking party status and admission of those contentions under §2.309. DPS has not waived its rights under 42 U.S.C. §2021(l) or 10 CFR §315(c).[2]

RELEVANT LEGAL STANDARDS

Applicant and NRC Staff devote several pages to the proposition that a contention must meet strict tests before it can be admitted. DPS acknowledges there are strict standards for admissibility of contentions. However, Applicant and NRC Staff go far beyond those standards in their Answers. Because their arguments are at odds with the plain language of the regulations, we believe a brief review of the relevant provisions is necessary.

The issue which the Board is asked to resolve at this stage, other than standing, which is

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[2] The argument advanced here is virtually identical to one advanced by DPS in the Uprate proceeding. It is reiterated here to preserve the position stated there.

not an issue with respect to DPS, is whether one or more contentions offered by DPS are admissible. "[T]he Atomic Safety and Licensing Board designated to rule on the . . . petition for leave to intervene will grant the . . . petition if it determines that the . . . petitioner has standing . . . and has proposed at least one admissible contention that meets the requirements of paragraph (f) of this section." 10 CFR §2.309 (a). Subparagraph (f) requires the contention be set forth with "particularity", that it contain a "specific statement of the issue of law or fact to be raised or controverted" and that it be accompanied by a number of other items, including "a brief explanation of the basis for the contention * * * a concise statement of the alleged facts or expert opinions which support the . . . petitioner's position on the issue * * * [and] sufficient information to show that a genuine dispute exists with the applicant . . . on a material issue of fact or law." 10 CFR §2.309 (f)(1). Nothing in Part 2 provides any support for the arguments advanced by Applicant and NRC Staff that the Board is to address the merits of the contentions in deciding whether they are admissible.

At no point in the regulations is the petitioner required to plead the bases or the supporting evidence with specificity[3] or to provide a listing of all the bases or supporting evidence upon which petitioner intends to rely in the hearing. In fact, by requiring only that the statement of bases be "brief" and the supporting evidence be "concise" the regulations clearly contemplate much more will be presented once the contention is admitted. The Commission has been clear that while a genuine dispute warranting a hearing must be shown, a petitioner does not have to prove its contention at the pleading stage. *In the Matter of Private Fuel Storage L.L.C,*

[3] Specific reference is to be made to the portions of the application which petitioner disputes. 10 CFR §2.309 (f)(1)(vi). As discussed below, DPS has done that.

(Independent Spent Fuel Storage Installation) NRC Docket No. 72-22-ISFSI CLO-04-22 at 8 (August 17, 2004), 2004 WL 2049726 (NRC). The issue now is whether the contention is admissible.

The Answers also treat each basis and each piece of supporting evidence offered as to each contention as a separate statement in support of the contention, arguing that because a particular basis or particular piece of evidence, standing alone, does not adequately support the contention, the contention should be rejected. The regulations contemplate exactly the opposite result. Contention admissibility is to be judged by the totality of factors listed in §2.309(f)(1). In addition, because it is the contention which is admitted, not the bases or the supporting evidence, it follows that once a contention is admitted, the contention may be supported by any evidence and bases[4]. Other parties are free to challenge those bases and evidence when offered under the rules applicable to the form of hearing granted, but not to seek or obtain a preliminary ruling at the contentions admissibility stage on the admissibility of any basis or supporting evidence with regard to an admitted contention.

Applicant and the Staff base many of their arguments on the assumption that the Board is to resolve disputes over the evidence in deciding whether to admit a contention.Such an approach

[4] "There is no regulatory requirement that an intervenor supply all the bases known at the time he files a contention. What is required is the filing of bases that the intervenor intends to rely on." The question in determining whether to admit a new basis for an already admitted contention is whether it is timely to consider the new basis, in light of its seriousness and of the timeliness with which it has been raised. "The more serious an issue, the more important it is for this Board to consider it. We can, indeed, always determine that a serious issue that falls within the scope of an admitted contention must be considered in order to assemble an adequate record." *In the Matter of Georgia Power Company* (Vogtle Electric Generating Plant, Units 1 and 2), 40 N.R.C. 37 at 2, 1994 WL 612194 (NRC) (July 1994)

to admissibility of contentions is directly contrary to the language of the regulations and makes no sense. The regulations require that petitioner submit "sufficient information to show that a genuine dispute exists with the applicant . . . on a material issue of law or fact." If the Board were required to resolve these disputes, there would be no need for the summary judgment procedures contained in Subparts G and L. In addition, since the contentions are presented before any discovery, even the mandatory discovery provided by 10 CFR §2.336, and the regulations require only a "brief explanation" of the bases and a "concise statement" of the supporting evidence, it would be unreasonable to expect petitioner to be in a position to present all the reasoning in support of each basis and all the evidence which demonstrates why a contention and its bases are factually correct.[5] If such a requirement existed, the Board would end up holding an evidentiary hearing to determine whether it should hold an evidentiary hearing. Clearly the regulations do not require such an absurd result. The regulations contemplate something more in the nature of a proffer of evidence that, if proven correct, would support the contention and bases.[6]

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[5] At a minimum, if the contention admissibility stage of the proceeding were a summary judgment proceeding, the protections provided by the summary judgment procedures should be followed including that 1) evidence offered by way of affidavit would have to be countered by contrary affidavit evidence, 2) where additional discovery was needed to properly respond to the motion an opportunity for such discovery would be provided and 3) all inferences that can be drawn from the evidence would be in favor of the party opposing summary judgment. Since neither Applicant nor the Staff have countered the affidavit evidence offered by DPS, they can not possibly prevail under a summary judgment standard.

[6]The Staff cites to theYankee Rowe case for the proposition that the Board is required to determine whose view of the facts is correct in passing on the admissibility of contentions. The case does not support such an extreme view as evidenced by the quoted language in the Staff's brief. All the Board in that case is recognizing is that in evaluating whether the particular evidence upon which petitioner relies actually supports the proposition for which it is cited the Board should look at the entire document. That is much different than the proposition offered by the Staff which is that the Board should not only look at the document relied upon by petitioner but on any other documents which, according to the Staff, would support the opposite conclusion

The Answers submitted by Applicant and the Staff, to the extent they argue, by reference to other documents and other evidence, that DPS is in error in its factual assertions, actually demonstrate that the contention to which those facts are relevant is a contention as to which a genuine dispute as to material facts exists and thus the contention is admissible. The regulations would not have focused on the existence of a dispute as the basis for admissibility of a contention if it were the Board's task to resolve that dispute in deciding whether to admit a contention[7].

ARGUMENT

A. CONTENTIONS

Applicant and Staff give short shrift to the proper legal standard to apply in deciding whether to grant the admission of a contention, but then totally ignore that legal standard in their argument. We agree that "bald or conclusory allegations that . . . a dispute exists" are unacceptable and that the burden on the proponent of the contention is to "make a minimal

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from the one advanced by DPS.

[7]The ASLB has found that although an applicant put forth "extensive arguments that really go to the merits" of an issue which was the focus of a contention, that even though some of those arguments may prove to be meritorious, they are not grounds for rejecting those portions of a contention that the ASLB find to be admissible. The ASLB went on to find that despite Applicant's and the NRC Staff's claims to the contrary, Intervenor did support three parts of its contention with "expert opinion, documentary material, and with a reasonably specific explanation and fact-based argument sufficient to meet the requirements of the contention admissibility requirements in this regard." Additionally, the ASLB found that the parties differing on the meaning and import of various facts, statements from documents, and other evidence within the basis for the contention illustrated a "genuine dispute" rather than negating it, or any other requirement for an admissible contention. *In the Matter of Duke Energy Corporation,* (Catawba Nuclear Station, Units 1 and 2), Docket No's. 50-413-OLA, 50-414-OLA, ASLBP NO. 03-815-03-OLA, LBP-04-10 at 42 (April 2004), 2004 WL 1398219 (NRC).

showing that material facts are in dispute". We also agree that a contention should be rejected if there are "no facts to support [our] position and [if we] contemplate using discovery or cross-examination as a fishing expedition". 54 Fed. Reg. at 33,171. No fair reading of the DPS contentions could possible conclude that DPS failed to meet these standards or that DPS intends to use discovery or cross-examination to aimlessly look for evidence to support its contentions. The documents submitted in support of the contentions[8] and pages of meticulous discussion of the meaning of those documents, supported by and supplemented with the opinion of a highly qualified technical expert, belie any suggestion that DPS has made bald or conclusory assertions or that there are no facts which support its position. Between them Applicant and NRC Staff use dozens of pages just to attempt to demonstrate that the many bases and substantial supporting evidence offered by DPS are wrong but, significantly, they never charge that DPS failed to offer bases and evidence in support of its contentions, only that they disagree with the bases and the evidence.

[8] Although Applicant and the Staff make reference in their Answers to numerous documents, no documents are appended to their pleading and there is not even an affidavit attesting to the accuracy of the statements made about the documents. Neither DPS nor the Board has any basis on this record to test the accuracy of the assertions made or to scrutinize the document "both for what it does and does not show." *Yankee Atomic Elec. Co.* (Yankee Nuclear Power Station), LPB-96-2, 43 NRC 61, 90 *rev'd in part on other grounds*, CLI-96-7, 43 NRC 235 (1996). Opponents of intervention and contentions also have obligations that they must strictly meet and suffer the consequences of their failure to strictly comply with those obligations. Thus, DPS will oppose any attempt to amend the Answers or to introduce documents at the hearing on the admissibility of contentions. Nor is it relevant that some or even all of the documents may be publicly available. It is not appropriate to impose upon DPS or the Board the obligation of rounding up all the cited documents.

<u>First Contention</u> (Safety)

> **The Application must be denied because the Applicant has failed to provide the necessary information with regard to age management of primary containment concrete in accordance with 10 C.F.R. §54.21 such that the Commission cannot find that 10 C.F.R. §54.29(a) is met.**

Contention 1 identifies that the Applicant attempts to take credit in License Renewal Application (LRA) Section 3.5.2.2.1.3 for an exclusion from age management of the reduction of strength and modulus of concrete for primary containment concrete based on the general area of concrete not exceeding 150°F. Yet at page 5.2-8 of the updated final safety analysis report section (UFSAR), Application states the temperature in the normal ambient temperature in the drywell is about 135°F to 165°F. The Contention highlights this discrepancy, stating that the general area of the face of the concrete will be greater than 150°F when drywell temperature is at 165°F. Either Applicant must show the face of the general area of concrete is less than 150°F to invoke the exclusion, or must provide an age management program for reduction of strength and modulus of concrete for primary containment concrete. As a result of this discrepancy, NRC cannot make the finding required by 10 C.F.R. §54.29(a).

Both the Staff and the Applicant claim Contention 1 is inadmissible because it is vague and unsupported (Applicant Answer at 11) or speculative and conclusory (Staff Answer at 11), lacking an adequate basis, and that therefore it fails to demonstrate the existence of a genuine dispute concerning a material issue. *Id.*

1. Contention 1 is stated with particularity and specificity

Applicant claims, in its Answer at 12, that Contention 1 lacks particularity and specificity. However, that is not the case. A contention is understood and its reach hinges upon its terms

coupled with its stated bases. The bases clearly states:

> [T]he Applicant improperly excludes the attribute of *reduction of strength and modulus of the primary containment structure due to elevated temperature.* The Applicant claims this attribute is not an aging effect requiring management. However, the primary containment normal operating temperature limit is above the limit for excluding this attribute from consideration.

DPS Petition at 10. The aging effect at issue (see Applicant Answer at 12) is the attribute of *reduction of strength and modulus of the primary containment structure due to elevated temperature.* The aspect of the aging management program being challenged (*id.*) is the lack of an aging management program for the attribute of *reduction of strength and modulus of the primary containment structure due to elevated temperature.* The necessary information that is lacking (*id.*) is information to resolve the inconsistency regarding the containment normal temperature limit above the limit for excluding the attribute from aging management consideration. Applicant is not correct in this claim.

2. Use of a temperature of 165°F next to the steel wall of the drywell is not incorrect for determining the primary containment temperature of a significant general area of concrete outside the steel drywell

The crux of Contention 1 is that the UFSAR Section 5.2.3.2 states that the temperature in the drywell is about 135°F to 165°F, and that using the higher temperature, 165°F, results in general area temperatures for concrete of greater than 150°F. The Applicant identifies a statement from UFSAR Section 5.2.3.7 that states that the drywell is cooled by four cooling units, which maintain an average temperature of 150°F, with a maximum of 135°F in vicinity of the recirculating pump motors. *Id.* at 13. The Applicant then incorrectly states, "DPS' reference to the UFSAR is selective and ignores the statements indicating that the general temperature in the drywell is 150°F, consistent with the section of the ASME Code discussed in the Application." *Id.*

The Applicant is incorrect for the following reasons. Contention 1 refers to the concrete outside the steel drywell[9]. If the area near the vicinity of the recirculating pump motors is maintained with a maximum of 135°F, the average is 150°F, and the peak is as high as 165°F, it follows that some portion of the area away from the recirculating pump motors is at 165°F. Since the recirculating pump motors are toward the inside of the drywell near the reactor, the area away from recirculating pump motors are the outer walls of the drywell. It is the temperature on the outer walls that controls the heat transfer and gradient through the walls to the concrete outside the steel drywell.

Further regarding the use of 165 °F for a general area temperature for concrete surface temperature determination, we are providing a portion of *Vermont Yankee Summary Report of Plant Environmental Conditions for Environmental Qualification Program,* Rev. 0, March 19, 1984, as Vermont Reply Exhibit 2. This includes page 4, "Normal Operating Plant Environments," which includes drywell operating temperatures. This information was developed from actual thermocouple readings[10]. This information shows that, for the general area from El. 270 ft to El. 315 ft, an average temperature of 185 °F should be used.. This is an average temperature applicable for use for the general area, as opposed to the peak temperature, listed as

[9] See NRC Staff Answer at 10-11 which correctly states that the scope of this contention is limited to the strength and modulus of the primary containment structure, and refers to the concrete outside the steel drywell.

[10] While this information is likely not current, at least it is representative of the thermal operation of the drywell that affected drywell concrete properties during a period of operation in 1980's. It is the temperature history that is relevant in the consideration of the attribute of *reduction of strength and modulus of the primary containment structure due to elevated temperature.* Therefore, this data from 1984 is specifically relevant. In addition, the temperature measurements were made when VY's maximum operation was 100% of thermal power. It now is allowed to operate at 120% of thermal power.

195 °F., which would be applicable for local area usage. This information from Vermont Reply Exhibit 2 demonstrates it is not incorrect to use 165 °F for the temperature next to the steel wall inside the drywell for determining the general area temperature for primary containment concrete outside the steel drywell between El. 270 ft. and El. 315 ft[11].

3. The expert statements of Mr. Sherman regarding concrete temperature are reliable

Both NRC Staff (Answer at 11-12) and Applicant (Answer at 13-14) claim that Mr. Sherman's statement, Declaration for Petition at ¶8, ("the concrete surface behind the steel shell will closely match the drywell ambient temperature,") is not sufficient basis within NRC rules and precedents to demonstrate the existence of a genuine dispute on a material issue of law or fact. That assertion from Applicant and NRC Staff is not supported by any expert affidavit and is itself a "conclusory and speculative assertion." Examination of Mr. Sherman's resume shows he is a registered professional engineer. His area of professional expertise is mechanical engineering, and heat transfer is a direct component of professional mechanical engineer expertise. Given a heat transfer scenario with an inside temperature of 165 °F, a 2.5-inch steel plate, a 2-inch sand gap, a six foot concrete wall, and an outside temperature of 100 °F, it is a reasonable conclusion, from a qualified expert that "the concrete surface behind the steel shell will closely match the drywell ambient temperature."

It is telling that neither Applicant nor NRC Staff have offered an expert opinion to refute the conclusion presented by Mr. Sherman. This is particularly significant because the calculation

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[11]The Applicant's statement in its Answer at 13 accusing DPS of selectively using statements and ignoring others is disturbing. We have shown that the accusation is wrong, and Entergy should have known that it was wrong. We note that none of the Applicant's factual assertions are supported by an affidavit of a credible expert.

required is basic heat transfer science and easily done. As the calculation described below demonstrates, the likely reason neither Applicant nor NRC Staff did a calculation is that their experts told them what Mr. Sherman had already concluded - 2.5-inches of steel and a 2-inch sand gap would not be sufficient to reduce 165 °F to 150 °F at the concrete wall.

Mr. Sherman's statement is correct and a sufficient basis to demonstrate the existence of a genuine dispute on a material issue of law or fact, and so it should be considered. Provision of actual heat transfer calculations are a level of detail that should be reserved for the evidence of the hearing and not an initial petition. Nevertheless, Mr. Sherman has prepared a calculation to demonstrate the accuracy of his statement at ¶8 of his Declaration for Petition.

The sample heat transfer calculation is for a representative cross section at El. 280 ft through the drywell to assess the temperature on the face of the concrete outside the steel drywell. *Marks' Standard Handbook for Mechanical Engineers*, Eighth Edition, 1978, McGraw Hill, pp. 4-59 to 4-70 (Transmission of Heat by Conduction and Convection) is used for the calculation. Data for the calculation was taken from Entergy's *License Renewal Application, Amendment No. 2*, dated May 15, 2006 (Vermont Reply Exhibit 1). This submittal identifies that, above the transition zone from spherical to cylindrical portions, the drywell is separated from reinforced concrete by a two-inch gap. The gap below this transition is filled with sand. In addition, the Amendment refers to the nominal plate thickness of the drywell as 2.5 inches.

The calculation assumes a steel plate of 2.5 inches, a sand-filled gap of 2 inches, and a concrete thickness of 6 feet, with drywell temperature at 165°F, the maximum value from UFSAR Section 5.2.3.2, and a reactor building temperature of 100°F. It was assumed that the drywell (near the drywell shell) and the reactor building were at their respective temperatures long enough

such that the steel surface inside the drywell and the concrete surface temperature in the reactor building were at these respective temperatures. The following thermal conductivities, in units of btu/hr/ft^2/°F/ft, were taken from the *Marks Handbook:* steel plate - 26.2, dry sand - 0.188, concrete - 1.05.

At equilibrium, the results of this temperature gradient are:

Temperature at steel surface in the drywell - 165°F

Temperature at the steel/sand interface - 164.9°F

Temperature at the inside concrete face - 156.2°F

In this calculation, approximately 8 inches of thickness of the concrete remains over 150°F. This calculation confirms Mr. Sherman's statement that "the concrete surface behind the steel shell will closely match the drywell ambient temperature."

The foregoing has demonstrated that Vermont Contention 1 has an adequate and sufficient basis, and a genuine dispute exists concerning a material issue. Contention 1 should be admitted.

<u>Second Contention</u> (Environmental)

> **The Application must be denied because Applicant has failed to comply with the requirements of 10 CFR §51.53(c)(3)(iv) by failing to include new and significant information regarding the substantial likelihood that spent fuel will have to be stored at the Vermont Yankee site longer than evaluated in the GEIS and perhaps indefinitely and thus has failed to provide the necessary environmental information with regard to onsite land use in accordance with 10 C.F.R. §54.23 such that the Commission cannot find that the applicable requirements of Subpart A of 10 C.F.R. Part 50 have been satisfied (10 C.F.R. §54.29(b)).**

INTRODUCTION

The central thesis of the arguments advanced by NRC Staff and Entergy in opposition to DPS Contention 2 is that the Commission has, contrary to all reason and in contravention of well-established legal principles, declared that no intervenor may ever present for consideration by an ASLB the issue of whether "new and significant information not considered in the GEIS analysis" exists thus warranting further analysis of those issues.[12] DPS rejects this ungenerous view of the Commission's intent in adopting the GEIS and, as the following analysis amply demonstrates, the Commission also rejects such a rigid and unreasoned position.

The error in the arguments advanced by NRC Staff and Entergy begins with their misinterpretation of DPS Contention 2 which reads as follows:

> The Application must be denied because Applicant has failed to comply with the requirements of 10 CFR §51.53(c)(3)(iv) by failing to include new and significant information regarding the substantial likelihood that spent fuel will have to be

[12] NRC Staff takes the wholly indefensible position that "[e]ven if there was [sic] new and significant information regarding the long-term storage of high-level waste beyond the period of license renewal, it would not need to be included in the GEIS, or any supplement thereto, as it is beyond [sic] scope." Staff Ans. at 17. Thus, NRC Staff urges the Board to adopt the absurd view that evidence, no matter how compelling, that the GEIS is incorrect should be ignored because the GEIS is, by rule, declared to be correct. Clearly, the Commission intends no such result. *See* 61 FR at 28471

> stored at the Vermont Yankee site longer than evaluated in the GEIS and perhaps indefinitely and thus has failed to provide the necessary environmental information with regard to onsite land use in accordance with 10 C.F.R. §54.23 such that the Commission cannot find that the applicable requirements of Subpart A of 10 C.F.R. Part 50 have been satisfied (10 C.F.R. §54.29(b))

First, the focus of the contention, as it must be at this stage, is the failure of the Applicant to provide certain information required to be provided.[13] Second, there is no dispute that Applicant has failed to provide that information. There is no discussion in the Application of the use of the site for spent fuel storage for an indefinite period beyond the license renewal date of 2032, much less the environmental impacts of such use. Third, the real issue at this stage of the proceeding is whether Applicant is legally required to provide such new and significant information regarding on-site land use. Fourth, the focus of the contention is the additional spent fuel that will be

[13] As noted in the Petition, failures by NRC Staff cannot be the subject of a contention at this time because NRC Staff has yet to publish a draft, much less, a final impact statement. Contrary to the assertion by Entergy, NRC Staff failures to comply with NEPA obligations can be the basis for a license denial since it is the major federal action by the NRC, whether to approve or disapprove the proposed license extension, to which NEPA is directed and it is the NRC Staff that has been delegated the responsibility to engage in the necessary review and analysis to demonstrate compliance with NEPA. Unlike the AEA, NEPA does not impose substantive requirements but rather process requirements on the NRC. NRC is required to engage in a certain process which includes consideration of relevant evidence and fully addressing that evidence in reaching conclusions. Ultimately it is the Board which decides whether the NEPA process has been fulfilled. Ignoring new and significant evidence that may alter prior conclusions on potential environmental impacts is the kind of procedural error which NEPA prohibits. If NRC ignores new and significant information, as Applicant and NRC Staff urge be done, NEPA compliance will be insufficient and the major federal action NRC is about to take in deciding whether to grant a 20 year license extension will be null and void. Thus, NRC compliance with NEPA is necessarily an issue before this Board. When, as here, Applicant fails to meet its obligations to provide information required by NRC Regulations, it is the regulations, not NEPA, that form the basis for the license denial. *See* 10 CFR §2.309)f)(vi)("if the petitioner believes that the application fails to contain information on a relevant matter as required by law, the identification of each failure and the supporting reasons for the petitioner's belief"). Contention 2 clearly meets this contention filing requirement.

generated after 2012 if VY is given a license extension. As the initial Petition demonstrates, it is the added burden that will be placed on the reasonably anticipated off-site waste disposal options, which options are insufficient to handle the post-2012 spent fuel, that will create the indefinite storage of spent fuel at the VY site.

Both Applicant and NRC Staff assume, erroneously, that this contention is about waste confidence. It is not. The waste confidence proceeding does not directly address the issue of the environmental impact on land use and state resources of the indefinite storage of spent fuel at the site of a nuclear reactor after the time when the reactor is no longer operating. Rather, it focuses on the radiological and other risk impacts on the environment. Moreover, the GEIS for license extensions also does not address this issue. Thus, it is irrelevant that 20 years ago the Commission decided that, based on the information then available to it, there was reasonable assurance that radiation and other risks would not endanger public health and safety or the environment and that there would be a place to store the spent fuel after each nuclear power plant had closed.

In addition, even if the issue of indefinite spent fuel storage after plant shut down were addressed years ago, the issue raised by Contention 2 is the failure of the Applicant to provide new and significant information regarding the likelihood that it will need to use the VY site for spent fuel storage for much longer than previously assumed and to evaluate the environmental impact of such longer use. For the sake of argument, we assume the Commission had a substantial evidentiary basis in 1984 and 1990 to conclude that spent fuel could be stored safely at or away from the reactor site for 30 years after the license, including any extended license, had expired. We also assume a substantial evidentiary basis existed in1990 to conclude that "at least

one mined geologic repository will be available within the first quarter of the twenty-first century, and sufficient repository capacity will be available within 30 years beyond the licensed life for operation of any reactor to dispose of the commercial high-level waste and spent fuel originating in such reactor and generated up to that time." However, no where in the waste confidence finding does the Commission discuss or purport to discuss the separate question of the environmental impact of the indefinite use of the reactor site for spent fuel storage.[14]
As noted in Contention 2, it is that continued site use that raises environmental concerns, a concern that transcends the current assumptions regarding the separate issue of whether extended on-site storage of spent fuel will be safe or available. There is substantial new and significant information that has emerged on these issues that is not discussed in the GEIS.

In its wisdom, the Commission has provided several mechanisms by which such new and significant information can be brought to its attention. One mechanism by which the Commission sought to be assured that it was aware of any new and significant information was to impose on the Applicant the obligation to report to it, in the Environmental Report, any new or significant information regarding potential environmental impacts not previously evaluated. 10 CFR §51.53(c)(3)(iv). Because the Applicant has failed to meet this obligation, one remedy allowed by the rules is for a prospective intervenor to challenge the failure of an applicant to, on its own, provide the new and significant information in its application. 10 CFR §2.309(f)(vi). DPS

[14] Significantly, when the Commission discusses the environmental impact of extended storage of spent fuel it is in the context of the risk to the environment - i.e. the radiological and other consequences of long term storage - not the land use questions. That analysis underlies the conclusions stated in 10 CFR §51.23 that there is no significant environmental impact associated with storage of spent fuel at the reactor site for 30 years after reactor shutdown, a conclusion which does not address either land use or indefinite spent fuel storage.

Contention 2 is based on the failure of Entergy to make the required disclosures.

There is a dispute, one worthy of consideration by the Board, as to whether the information is in fact "new and significant" within the meaning of the supplementation rules. Resolution of that issue with a finding that the information is "new and significant" may then warrant further proceedings either by the Board or the Commission to integrate that new and significant information into the environmental analysis of the proposed extension. Until that integration has occurred and the impacts of indefinite storage of spent fuel at VY have been quantified, it will not be possible to complete the environmental review for the VY extension and to reach a final decision on the proposed action.

ARGUMENT

A. ENTERGY HAS A DUTY TO IDENTIFY NEW AND SIGNIFICANT INFORMATION

The critical regulatory standard at issue is 10 CFR §51.53(c)(3)(iv) which provides in pertinent part:

> The environmental report must contain any new and significant information regarding the environmental impacts of license renewal of which the applicant is aware.

Entergy is aware of new and significant information regarding the environmental impact of land use from spent fuel storage and is aware that it may have to keep such spent fuel on site for a much longer time than assumed in the GEIS. In *Entergy Nuclear Generating Co. v. U.S.*, 64 Fed.Cl. 336 (2005) Entergy successfully sued the United States on the theory that DOE had breached a contractual duty to take possession of, and title to, spent nuclear fuel (SNF) within 63 months after a utility submitted a delivery commitment schedule (DCS) with regard to such SNF.

In that suit, and at the urging of Entergy, the Court of Claims, in reliance on the stipulation of the parties and otherwise undisputed facts reached the following conclusion:

> This aborted effort in 2004 to reinstitute the DCS process signals that no disposal of SNF will occur during 2010, taking into account the 63-month period between designation and collection, and moreover that disposal may not occur within any foreseeable time in the future. No repository is available.

Id. 64 Fed.Cl. at 340 (citation omitted)(the chaotic nature of the entire spent fuel storage management scheme is detailed in the Court's opinion at footnotes 3 and 4). Entergy was fully capable of setting forth these new and significant facts, plus we suspect much more information not readily available from the printed case, in order to meet its obligations under 10 CFR §51.53(c)(3)(iv) but failed to do so, thus depriving the NRC, potential intervenors, and this Board of the truth about the uncertainty in how Entergy will manage the spent fuel it proposes to generate over the extended 20 years of operation of VY.

Once before VY ignored the risk that it might produce nuclear waste for which no disposal remedy existed and now finds itself left holding that spent fuel indefinitely without a viable solution in sight. *See Vermont Yankee Nuclear Power Corp. v. Natural Resources Defense Council*, 435 U.S. 519 (1978). It is imprudent for VY to fail to heed the lesson it should have learned from its prior decision to proceed to produce nuclear waste when no waste disposal solution was available and fail to disclose in its ER that once again it proposes to generate years of nuclear waste without any assurance that when the time comes it will have any place to keep that waste except at the plant site. This Board has the right to consider the potential impact, on the 125 acres of land owned by VY and the thousands of acres of nearby land, of indefinite storage of spent nuclear fuel at VY.

Neither NRC Staff nor Applicant deny the profound potential impacts on local land use and Vermont State resources that can occur if spent fuel remains at the site indefinitely following closure of the reactor. Nor do they challenge the fact that at present there is substantial information that storage of spent fuel at VY may well extend beyond any date assumed in the GEIS. Nor do they challenge the fact that based on the currently available information there is no reliable basis to conclude that the additional fuel to be generated by extended operation will be able to be stored off-site at any time in the foreseeable future. Even the Commission concedes that there is no basis to assume that sufficient storage will be available for all the fuel to be generated if all plants obtain license extensions. *See* GEIS §6.4.6.2.

Rather, NRC Staff and Applicant seek to obfuscate the real issue by citing to the Commission's decision, codified in 10 CFR §51.23, that the Commission is confident that within 30 years after the shutdown of all nuclear plants a waste disposal solution will be found that can safely store nuclear wastes and that in the interim the risks to human health and the environment from extended storage of nuclear fuel at nuclear reactor sites will be acceptable.

To fully understand what the Commission did and did not do in adopting 10 CFR §51.23 it is necessary to examine the statement of considerations published by it and in particular its discussion of the issue of non-radiological impacts of extended spent fuel storage at the reactor site. In the extended discussion of comments relating to the non-radiological impacts of spent fuel storage at the reactor cite, the Commission wrote:

> B. Non-Radiological Consequences of Spent Fuel Storage
>
> The Commission's fourth finding rested in part on the Commission's determination that there are no significant non-radiological consequences due to the extended storage of spent fuel which could adversely affect the environment. The public was invited to comment also on this finding and to provide a detailed discussion of

> any such environmental impacts. Mr. Marvin Lewis asserted that the continuous storage of spent fuel under water for 30 years or more requires unprecedented institutional guarantees. He also noted that there had been no consideration of financial, economic and security implications of storage for 30 or more years. Mr. Lewis did not expand upon these assertions to explain how they would result in significant non-radiological environmental consequences. In any event, the more than twenty years of experience with storing spent fuel demonstrates that storage of spent fuel for 30 years or more does not require unprecedented institutional guarantees or raise unique questions regarding finances, economics or the security of extended spent fuel storage. Further, the Commission will require all reactor licensees, 5 years before expiration of their operating license to provide a plan for managing the spent fuel prior to disposal. Moreover, the record documents referred to by UNWMG-EEI, DOE and AIF show that there are no significant non-radiological environmental impacts associated with the extended storage of spent fuels. The amount of heat given off by spent fuel decreases with time as the fuel ages and decays radioactively. No additional land needs to be devoted to storage facilities because reactor sites have adequate space for additional spent fuel pools or dry storage installations. The additional energy and water needed to maintain spent fuel storage is also environmentally insignificant. No commentor has challenged these assessments of environmental impacts and the Commission has no reason to question their validity. Under these circumstances, the Commission has no reason to reassess its prior determination that extended storage of spent fuel will present no significant non-radiological consequences which could adversely affect the environment.

49 FR 34658, 34665. Thus, there was no consideration of the type of land use impacts that DPS addresses in Contention 2. Those land use impacts are inherently site specific and need to be considered in the context of this site.

But, NRC Staff and Applicant assert, the NRC has already addressed the land use impacts and found them subject to resolution generically and too small to be worthy of further analysis. However, an exploration of the Statement of Considerations for 10 CFR §51.53 reveals that NRC did not address the site specific concerns raised here by DPS since it merely relied on the previous waste confidence findings.

> Response. As stated at 61 FR 28477, the Commission acknowledges that there is uncertainty in the schedule of availability of disposal facilities for LLW

> and HLW. The Commission understands the continuing concern of the States and of the public over the prospects for timely development of waste disposal facilities. The uncertainty in the schedule of availability of disposal facilities is especially of concern because of the waste currently being generated during the initial licensing term of power reactors. <u>The Commission, however, continues to believe that there is sufficient understanding of and experience with the storage of LLW and HLW to conclude that the waste generated at any plant as a result of license renewal can be stored safely and without significant environmental impacts prior to permanent disposal. The Commission believes that conditioning individual license renewal decisions on resolution of radioactive waste disposal issues is not warranted because the Commission has already made a generic determination, codified in 10 CFR 51.23, that spent fuel generated at any reactor can be stored safely and without significant environmental impacts for at least 30 years beyond a license renewal term and that there will be a repository available within the first quarter of the twenty-first century.</u> The waste confidence decision is discussed in Chapter 6 of NUREG-1437, "Generic Environmental Impact Statement for License Renewal for Nuclear Plants," May 1996. The Commission similarly believes that enough is known regarding the effects of permanent disposal to reach the generic conclusion in the rule. The rule is not based on the assumption that Yucca Mountain will be licensed. <u>Also from a regulatory policy perspective, the Commission disagrees with the view of one state that each renewal applicant should come forward with an analysis of the HLW storage and disposal environmental effects. This is a national problem of essentially the same degree of complexity and uncertainty for every renewal application and it would not be useful to have a repetitive reconsideration of the matter.</u>
>
> <u>The Commission further believes that the provisions in the present rule and elsewhere in the Commission's regulations adequately provide for the introduction and consideration of new significant information in license renewal reviews</u>, and that the 10 year review cycle for the rule and the GEIS adequately provides for Commission reassessment of the status of LLW and HLW disposal programs. The Commission recognizes that the possibility of significant unexpected events remains open. Consequently, the Commission will review its conclusions on these waste findings should significant and pertinent unexpected events occur (see also, 49 FR 34658 (August 31, 1984)). In view of the Commission's favorable conclusions regarding prospects for safe and environmentally acceptable waste disposal, it sees no need for conditioning licenses as recommended. The Category 1 designations for these three issues [low-level waste storage and disposal, offsite radiological impacts (spent fuel and high-level waste disposal), and on-site spent fuel] in the final rule has not been changed in response to these comments.

61 FR 66537, 66538 (emphasis added). Thus, it can be seen that the focus of the Commission analysis was on the generic issue of whether plants should be licensed at all in the face of waste

disposal uncertainty, on the generic issue of the potential radiological and other risk and their impact on the environment from storage of spent fuel at the reactor site and not the site specific question of whether indefinite storage of spent fuel at the reactor site would cause a significant environmental impact on land use at that site.

This analysis of the reach and intent of 10 CFR §51.23 is confirmed by an analysis of the GEIS, which is the place where the Commission sought to address the conventional environmental impacts of license extensions, as opposed to radiological and other risks. The GEIS assigns a Category 1, i.e. not requiring individual analysis, to the on-site storage of the additional spent fuel generated by an extended license. 10 CFR Part 51, Subpart A, Appendix B. But the statement of considerations makes clear that this classification is not intended to address the site specific issue raised by DPS Contention 2:

> Thus, continued storage of spent fuel on site may be an issue for some utilities regardless of their license renewal plans. GEIS, §6.4.6.1
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> ***
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> Under the Waste Confidence Rule, NRC has determined that spent fuel can be stored on-site for at least 30 years beyond the licensed (and license renewal) operating life of nuclear power plants safely and with minimal environmental impact (54 FR 39765; 55 FR 38472). This decision does not address the environmental impacts of storage during the additional 20 years of operation after license renewal. The additional spent fuel generated during this 20-year period poses three potential issues.
>
> First, under the Nuclear Waste Policy Act of 1982 (NWPA) as amended, DOE is authorized to dispose of up to 70,000 metric tonnes of heavy metal (MTHM) in the first repository before granting a construction authorization for a second. Under existing licenses, projected spent-fuel generation could exceed 70,000 MTHM as early as the year 2010. Possible extensions or renewals of operating licenses also need to be considered in assessing the need for and scheduling the second repository. It now appears that unless Congress lifts the capacity limit on the first repository--and unless this repository has the physical capacity to dispose of all spent fuel generated under both the original and extended or renewed licenses--it

> will be necessary to have at least one additional repository. Assuming that the first repository is available by 2025 and has a capacity on the order of 70,000 MTHM, additional disposal capacity would probably not be needed before about the year 2040 to avoid storing spent fuel at a reactor for more than 30 years after expiration of reactor operating licenses. GEIS, §6.4.6.2
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> ***
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> The Commission's waste confidence finding at 10 CFR 51.23 leaves only the on-site storage of spent fuel during the term of plant operation as a high-level-waste storage and disposal issue at the time of license renewal. The Commission's regulatory requirements and the experience with on-site storage of spent fuel in fuel pools and dry storage has been reviewed. <u>Within the context of a license renewal review and determination, the Commission finds that there is ample basis to conclude that continued storage of existing spent fuel and storage of spent fuel generated during the license renewal period can be accomplished safely and without significant environmental impacts. Radiological impacts will be well within regulatory limits; thus radiological impacts of on-site storage meet the standard for a conclusion of small impact. The nonradiological environmental impacts have been shown to be not significant; thus they are classified as small. The overall conclusion for on-site storage of spent fuel during the term of a renewed license is that the environmental impacts will be small for each plant</u>. The need for the consideration of mitigation alternatives within the context of renewal of a power reactor license has been considered, and the Commission concludes that its regulatory requirements already in place provide adequate mitigation incentives for on-site storage of spent fuel. On-site storage of spent fuel during the term of a renewed operating license is a Category 1 issue. GEIS §6.4.6.7 (Emphasis added)

In reviewing the entire portion of the GEIS addressing the spent fuel storage at the site following reactor shut down, there is no discussion of the issue of indefinite impacts on local land use and no discussion of the special nature of the land in the vicinity of any plant, much less VY. The entire discussion uses the concept of minimal environmental impact to refer to impacts on the environment from radiological and non-radiological risks from extended spent fuel storage.

In addition the S-3 Table, which is a major underpinning of the GEIS analysis, assumes that the reactor site will be safe and usable shortly after the plant has been decommissioned. *See* 61 FR 28467, 28479 ("Table S-3 does not take into account long-term onsite storage of LLW, mixed waste, and storage of spent fuel assemblies onsite for longer than 10 years, nor does it take

into account impacts from mixed waste disposal. The environmental impacts of these aspects of onsite storage are also addressed in Chapter 6 of the final GEIS and the findings are included in the final rule in Table B-1 of appendix B to 10 CFR part 51.").

DPS does not contest in this proceeding that the storage of the spent fuel at VY for an indefinite period beyond the date the reactor is shut down may be safe. But, the adjacent land and the VY land itself will not be able to be developed and used in the same manner as if the spent fuel had been removed from the site. It is that impact which has not been considered in S-3, the GEIS or anywhere else and it is that issue which needs to be addressed in this proceeding in order to fully characterize the potential environmental impact of the proposed license extension. The Applicant's failure to disclose information of which it was aware that bears on the issue on the duration of land use for spent fuel storage is a legitimate contention which may well lead to additional contentions once the draft and final EIS are presented by the Staff.

Because the challenge to Contention 2 by Applicant includes an attack on the merits of the statement that new and significant information exists, an attack which concedes this is an issue on which there is a material dispute of fact, we devote a few paragraphs to rebutting Applicant's unsupported assertions. We do not waive the basic argument that these factual disputes should be resolved in the hearing, not in the intervention process.

Applicant claims that the new and significant information identified by DPS was previously considered in the GEIS and is also not significant. Applicant Answer at 19-23. Applicant's claims are not correct.

While individual statements, quoted by the Applicant, were made in the Waste Confidence Decision about unexpected results at Yucca mountain, the possible need for a second repository,

and a possible reconsideration of reprocessing, it is the sum and combination of each of these occurring together which constitutes new and significant information. Also, Applicant ignores its own new and significant information regarding the unlikelihood that title to any spent fuel will be transferred from Applicant to the government and thus the unlikelihood that any of the spent fuel generated after 2012 will be transferred off-site in the foreseeable future. *See Entergy Nuclear Generating Co. v. U.S.*

There is no question that the discovery of groundwater at disposal levels at Yucca Mountain has created a compete paradigm shift. A primary reason Yucca Mountain was chosen was because it contained a unique geological formation that was thought to prevent groundwater intrusion. The fact that groundwater has been recently discovered and the paradigm for design has shifted is seen in the U.S Nuclear Waste Technical Review Board's (NWTRB's) *Report to the U.S. Congress and Secretary of Energy*, January 1, 2005 to February 28, 2006 ("NWTRB Report"). The executive summary contains the following:

> Two potentially significant natural barriers at Yucca Mountain—the unsaturated zone beneath the repository horizon and the saturated zone— can isolate radionuclides that might be released from the emplaced waste packages. The Board believes that the Project has made great strides over the last few years in developing a sound understanding of the magnitude and <u>rates of mountain-scale groundwater flow in the unsaturated and saturated zones under ambient temperatures and current climatic conditions</u>.

NWTRB Report at 1. (Emphasis added.)

> A key driver in the performance of the repository, both preclosure and postclosure, is temperature. The temperature of the spent nuclear fuel affects the integrity of the fuel cladding and the susceptibility of the waste-package material to localized or general corrosion. The temperature and time profiles in the near-field environment of the drift affect tunnel degradation, causing more fracture pathways, drift separation, <u>and movement of water or water vapor in the unsaturated zone.</u> How these temperatures are controlled is determined by the Project's thermal-management strategy, which identifies controlling criteria, including the maximum thermal loading of the waste packages, line loading in the emplacement drift, and

peak temperatures and zones for pillar separation.

Id. (Emphasis added.)

> The Board has concerns about the technical basis underlying the Project's thermal-management strategy. First, the technical basis for the Project's choice of thermal criteria to limit temperature is not well-defined. The Board believes that the Project should articulate in a transparent way the basis for its thermal criteria. Second, the implications for thermal management of the Project's provisional decision to develop and implement a standardized canister for storing, transporting, and disposing of spent nuclear fuel do not seem to have been evaluated fully. The Board is particularly concerned about the ability of the utilities to blend the spent nuclear fuel to the required thermal loading, given the spent nuclear fuel available in the spent-fuel pools, the increasing volume of spent nuclear fuel in dry storage at reactors, and the trend toward higher burn-up fuel. Moreover, the Board is concerned that the constraints imposed by line-load requirements during emplacement have not been fully represented or understood in terms of surface facility design and operation. Third, the Board is not persuaded that the thermal-hydrologic models being used to predict postclosure temperature, relative humidity, and vapor transport within the drifts have a strong technical basis.

*Id.*at 1,2. (Emphasis added.)

> The engineered barrier system consists of the spent nuclear fuel, including the cladding and the fuel pellets; the waste package, including any canister or basket holding the spent nuclear fuel or high-level radioactive waste; the waste package invert; the drip shield; and the backfill, if any. As do the natural barriers, the engineered barrier system can contribute to waste isolation.

Id. at 2,

> The Alloy-22 outer barrier of the waste package will not corrode significantly unless liquid water is present on the waste package surface. The higher the temperature at which liquid water is present, the greater is the concern, because metals generally corrode faster at higher temperatures and the susceptibility of metals to corrosion generally increases at higher temperatures. Project scientists have determined that dusts from ventilation air during the preclosure period would settle on waste package surfaces and would contain salts that could form saturated brines with boiling points on the order of 200°C.

Id. (Emphasis added.)

> The Project maintains that potential localized corrosion of Alloy-22 at elevated temperatures can be excluded from its performance-assessment calculations. The Board believes that the

> technical basis for the exclusion is not compelling, partly because only very limited corrosion data have been collected at temperatures above 150°C and partly because data showing cessation (stifling) of localized corrosion at lower temperatures may or may not be relevant to all conditions under which localized corrosion could occur in the proposed repository. The Board strongly urges the Project to continue collecting data that might justify its assumption that localized corrosion will not occur at temperatures as high as 200°

Id.

These statements from the executive summary of the NWTRB report illustrate that the project is now considering the presence of groundwater in its design. The body of the NWTRB Report is filled with details related to having to create a new design for the groundwater that has been discovered.

The change in national policy for waste disposal also constitutes new and significant standard. As stated in the petition, the Administration is embarked on a major new initiative labeled the Global Nuclear Energy Partnership (GNEP). As part of GNEP, the Administration proposes changing to a novel mode of reprocessing in which unused Uranium would be removed but Plutonium would remain in a form that does not promote proliferation of weapons-grade nuclear material. This novel mode of reprocessing is unproven.

Prominent political supporters of GNEP advocate retaining spent nuclear fuel in its present location while the nation embarks on a research program with an undefined schedule to try to find a reprocessing process that would meet these goals. They reason that, since spent fuel will not be disposed of, but rather reprocessed, it should not be moved until after it can be reprocessed. And further, the disposal plan at the repository would have to undergo a major modification to accept reprocessing waste forms instead of spent fuel. The result is that all spent fuel disposal plans would be on hold while it is determined if (not when) a reprocessing method could be developed.

Applicant is not correct regarding its comment that "the Commission explicitly

"recogniz[ed] the possibility" that the country might renew reprocessing of spent nuclear fuel. 55 Fed. Reg. at 38,489, 38,493."(Answer at 20). The referenced statements from 55 Fed. Reg. at 38,489, 38,493 applied to Waste Confidence Finding One pertaining to the technical feasibility of a repository, and not Finding Two which dealt with the schedule. There is no suggestion in the discussion for Finding Two that the Commission considered a return to reprocessing in its schedule determinations. And most pertinent, the Commission certainly did not envision a turn to a completely novel and unproven method of reprocessing with no set schedule and disposal plans on hold.

The Applicant gives short shrift (Answer at 21) to the changing political climate regarding spent nuclear fuel disposal. We believe this is wrong because we believe most involved with the spent fuel disposal dilemma would say it is primarily a political problem. Part of the new and significant information is the political landscape. We have an Administration, responsible for implementing spent fuel disposal, which is now promoting the novel GNEP. We have the most powerful nuclear advocate in the Senate, Sen. Domenici, also promoting GNEP and urging retention of spent fuel at their current locatoins. We have the most powerful Senator for the other party, Sen. Reid, as the primary opponent of Yucca Mountain development, also urging retention of spent fuel on their present sites indefinitely. This political landscape constitutes new and significant information which will have high impact on whether spent nuclear fuel will ever move, and the land use at Vermont Yankee.

Regarding a second repository, the Applicant quotes from Waste Confidence:

> The Commission also explicitly considered the first repository's capacity and the need for a second repository and concluded that "if the need for an additional repository is established, <u>Congress will provide the needed institutional support and funding, as it has for the first repository,</u>" and that it "need not at this time consider the institutional

> uncertainties arising from having to restart a second repository program." 55 Fed. Reg. at 38,502, 38,504.

Emphasis added. With the changed political landscape, there is no basis to believe "Congress will provide the needed institutional support and funding, as it has for the first repository."

The same is true regarding the GEIS statement about a second repository:

> Assuming that the first repository is available by 2025 and has a capacity on the order of 70,000 MTHM, additional disposal capacity would probably not be needed before about the year 2040 to avoid storing spent fuel at a reactor for more than 30 years after expiration of reactor operating licenses. GEIS, §6.4.6.2

If it took from 1985 until 2025, a period of 40 years, to develop the first repository, there is no basis to believe that a second repository, if started immediately, could be developed within 34 years, given the past history and present political landscape.

Finally, it is not each single item mentioned about that constitutes new and significant information, but it is the sum of all of these items that results in a situation where spent nuclear fuel will remain at Vermont indefinitely, creating a MODERATE to LARGE evaluation associated with this use of land.

B. NEW AND SIGNIFICANT INFORMATION REQUIRES A REVISION TO THE GEIS

In addition to the fact that the specific issue raised by DPS in Contention 2 is not addressed by the GEIS and therefore is not foreclosed from full consideration in this proceeding, there is another independent reason why Contention 2 is admissible. Assuming, as NRC Staff and Entergy argue, that the GEIS has addressed the issue of the use of land after the shut down of the reactor, has concluded that its use will be no longer than 30 years and that such use is a Category 1 impact, the Commission has explicitly provided that the question of whether there is new and significant information that would warrant amending the GEIS or ignoring its findings for a

specific case is a question which can be raised in this proceeding.

In the Statement of Consideration accompanying the adoption of amendments to 10 CFR Part 51, the Commission addressed the issue of how to deal with new and significant information in response to concerns from the public and many interested states. The Commission resolved the issue as follows:

> The major changes adopted as a result of these discussions are as follows:
>
> 1. The NRC will prepare a supplemental site-specific EIS, rather than an environmental assessment (as initially proposed), for each license renewal application. This SEIS will be a supplement to the GEIS. Additionally, the NRC will review comments on the draft SEIS and determine whether such comments introduce new and significant information not considered in the GEIS analysis. All comments on the applicability of the analyses of impacts codified in the rule and the analysis contained in the draft supplemental EIS will be addressed by NRC in the final supplemental EIS in accordance with 40 CFR 1503.4, regardless of whether the comment is directed to impacts in Category 1 or 2. Such comments will be addressed in the following manner:
>
> a. NRC's response to a comment regarding the applicability of the analysis of an impact codified in the rule to the plant in question may be a statement and explanation of its view that the analysis is adequate including, if applicable, consideration of the significance of new information. A commenter dissatisfied with such a response may file a petition for rulemaking under 10 CFR 2.802. If the commenter is successful in persuading the Commission that the new information does indicate that the analysis of an impact codified in the rule is incorrect in significant respects (either in general or with respect to the particular plant), a rulemaking proceeding will be initiated.
>
> b. If a commenter provides new information which is relevant to the plant and is also relevant to other plants (i.e., generic information) and that information demonstrates that the analysis of an impact codified in the final rule is incorrect, the NRC staff will seek Commission approval to either suspend the application of the rule on a generic basis with respect to the analysis or delay granting the renewal application (and possibly other renewal applications) until the analysis in the GEIS is updated and the rule amended. If the rule is suspended for the analysis, each supplemental EIS would reflect the corrected analysis until such time as the rule is amended.
>
> c. If a commenter provides new, site-specific information which demonstrates that

> the analysis of an impact codified in the rule is incorrect with respect to the particular plant, the NRC staff will seek Commission approval to waive the application of the rule with respect to that analysis in that specific renewal proceeding. The supplemental EIS would reflect the corrected analysis as appropriate.

61 FR 28467, 28470.

Step one in the process set forth by the Commission is the Applicant's ER submittal which is required to include any new and significant information. 10 CFR §51.53(c)(iv). The new and significant information requirement applies, as noted in the Statement of Considerations, to both Category 1 and Category 2 impacts. As noted in the Statement of Considerations the Staff also has an obligation with regard to receiving and considering new and significant information and seeking Commission approval for modifications in the GEIS in light of that information.[15] It is also incumbent upon the Board to consider whether new and significant information warrants consideration of additional environmental impacts not covered by the GEIS:

> (4) The supplemental environmental impact statement must contain the NRC staff's recommendation regarding the environmental acceptability of the license renewal action. <u>In order to make its recommendation and final conclusion on the proposed action</u>, the NRC staff, <u>adjudicatory officers</u>, and Commission <u>shall integrate the conclusions</u>, as amplified by the supporting information in the generic environmental impact statement for issues designated Category 1 (with the exception of offsite radiological impacts for collective effects and the disposal of spent fuel and high level waste) or resolved Category 2,information developed for

[15] DPS has provided NRC Staff with its views on the new and significant information addressed in Contention 2 by timely filing its comments in response to the Federal Register Notice, Vol 71, No. 77, Friday April 21, 2006, pages 20733-20735. June 23, 2006 Letter from William Sherman to Chief, Rules and Directives Branch. The staff is required to consider whether new and significant information warrants any change to the GEIS conclusions for the specific plant and include those in the Draft SEIS. 61 FR 28467, 28485 ("If the comments are determined to provide new and significant information bearing on the previous analysis in the GEIS, these comments will be considered and appropriately factored into the Commission's analysis in the SEIS. Public comments on the site-specific additional information provided by the applicant regarding Category 2 issues will be considered in the SEIS.").

> those open Category 2 issues applicable to the plant in accordance with §51.53(c)(3)(ii), and any significant new information. Given this information, the NRC staff, adjudicatory officers, and Commission shall determine whether or not the adverse environmental impacts of license renewal are so great that preserving the option of license renewal for energy planning decisionmakers would be unreasonable.

10 CFR §51.95(c)(4).[16] *See also* 10 CFR §51.104(a) delegating to the Board the task of resolving disagreements among the parties regarding the EIS findings in cases, like this, where an ASLB has been convened.

The regulations contemplate an iterative process with regard to new and significant information, beginning with Applicant's obligations under 10 CFR §51.53(c)(3)(iv). Thus a proposed intervenor must start the process of challenging the environmental impacts by challenging the Applicant's failure to identify new and significant information of which it is aware in either Category 1 or 2. Unless corrected, that failure alone would warrant denial of the proposed extension. If Applicant files all of the information of which it is aware that is new and significant regarding the duration of storage of spent fuel at VY following the expiration of the extended license, the focus will then shift to the NRC Staff and its obligations. In that event, however, the record will contain an admission from Applicant that new and significant information does exist. Applicant cannot avoid making this admission by the illegal expedient of failing to meet its obligations under 10 CFR §51.53(c)(iv).

If Applicant, in order to avoid such an admission, chooses to deny that there is any new and significant information, as it does here, then there is clear issue of disputed fact that the Board

[16] This section provides that site- specific environmental findings shall be amplified by GEIS findings ("with the exception of offsite radiological impacts for collective effects and the disposal of spent fuel and high level waste"), confirming that the portion of the GEIS that addresses spent fuel storage is focused on radiological environmental impacts, not land use.

is required to resolve. 10 CFR §2.309(f)(1)(vi). By presenting contrary evidence to that presented by DPS, Applicant and the Staff have conceded the contention does raise genuine factual disputes that warrant a hearing. *In the Matter of Duke Energy Corporation*, (Catawba Nuclear Station, Units 1 and 2), Docket No's. 50-413-OLA, 50-414-OLA, ASLBP NO. 03-815-03-OLA, LBP-04-10 at 42 (April 2004), 2004 WL 1398219 (NRC). To the extent the Board chooses to address the merits of the bases and supporting evidence offered by DPS, it is significant that only DPS provided any admissible evidence. The DPS factual submittal was attested to, by a qualified expert.[17] The contrary opinions, interpretations of documents and factual claims by Applicant and the Staff in opposition represent nothing more than the unsworn assertions of lawyers. Such lawyer assertions are insufficient to overcome the attested to evidence of DPS. *See* 10 CFR §2.710(b).

DPS will file contentions regarding the NRC Staff compliance with its obligations regarding new and significant information, assuming it is not in full compliance with its obligations, at the time of issuance of the Draft EIS in order to assure that the contentions are timely. It is not possible at this time to know what NRC Staff will do but there are four possibilities other than full compliance with the regulations and NEPA:

1. Fail to identify all the new and significant information;

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[17] Mr. Sherman is the Vermont State Nuclear Engineer, a position held since 1988. He brings special qualifications as an expert witness. He is briefed on nuclear matters by the NRC and attends numerous briefings by DOE and others regarding nuclear waste disposal planning. He is particularly well-qualified to offer his opinions on the factual subjects in dispute regarding admissibility of DPS contentions. Part of his responsibilities include following the activities of Vermont Yankee on a day-to-day basis. This provides for daily plant status notifications from Entergy and access to Vermont Yankee documents, many of which are reviewed at, but not taken from, the plant site. However, he does not have access to those internal Entergy documents in which Entergy assesses the likelihood that it will have to keep spent fuel at VY indefinitely.

2. Identify the information but deny that it is new and significant;

3. Identify the information, admit that it is new and significant but deny that it warrants any additional consideration of environmental impacts of the proposal;

4. Admit that modifications of the environmental impacts of the proposal are required but fail to properly identify and weigh those impacts.[18]

In this case, Applicant takes its stance at the most fundamental point by arguing that there is no new and significant information. However, whether it is correct in that belief is not a matter for resolution at the contention admissibility stage but goes to the merits of the contention and cannot be resolved until after the contention is admitted. Once the contention is admitted, Applicant will be obligated, under the disclosure provisions of 10 CFR §2.336(a)(2)(i) to identify and/or produce all documents "that are relevant to the contentions". This would include all the information on the likelihood that spent fuel will need to be stored at the VY for more than 30 years after VY operation ceases, whether or not Applicant believes it is new or significant.

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[18] If, as we believe is the case, the information that DPS has identified plus the additional information that should be supplied by Applicant demonstrates that there is a significant environmental impact which may be caused by granting the proposed extension and that the GEIS never addressed this issue, there is no need to amend the GEIS. The information will be site-specific - i.e. the environmental impact on land use in the area of the plant if the site is indefinitely used for spent fuel storage - and will be able to be fully analyzed in the SEIS. If, however, the Board concludes that the new and significant information addresses issues already covered by the GEIS, then the GEIS itself will need to be amended. The process for that, as it applies to the EIS outside the hearing process, is set forth in the Statement of Considerations quoted *supra*. 61 FR 28467, 28470. It involves the Staff making application to the Commission for a modification in the GEIS or the party aggrieved by the Staff refusal seeking an amendment of the rules. However, as noted in 10 CFR §51.104(a), where, as here, the EIS is issued in the context of an ASLB proceeding, the issue is resolved by the Board. What is unclear is whether the Board decides that the GEIS needs to be modified and then proceeds to do so or whether the Board recommends such action, essentially standing in the shoes of the NRC Staff, and the Commission makes the final decision. This issue may be somewhat academic since, whatever the Board does, the Commission will be the final arbiter.

Finally, there cannot be any doubt that the issue that underlies the DPS contention, i.e. whether there is new and significant information that would warrant modification of the environmental impacts as now apparently contemplated by Applicant and NRC Staff, is an appropriate issue for resolution in this hearing. In its Statement of Considerations accompanying the regulations governing the analysis of environmental impacts of proposed license extensions the Commission was careful to note that:

> The Commission will issue a final supplemental environmental impact statement for a license renewal application in accordance with 10 CFR 51.91 and 51.93 after considering the public comments related to new issues identified from the scoping and public comment process, Category 2 issues, <u>and any new and significant information regarding previously analyzed and codified Category 1 issues</u>.

61 FR 28467, 28485. Thus, the suggestion that DPS can only raise Contention 2 if it files a formal request pursuant to 10 CFR §2.335(b) ignores the extensive administrative history confirming that the Commission intends that claims of the existence of new and significant information warranting modifications to the GEIS are to be part of the SEIS and ASLB decision-making process. In addition, even if a §2.335(b) petition were required, the contention as filed, with the supporting affidavit of William Sherman, meets the requirements of the regulation. The Contention and affidavit identify the way in which the GEIS designation of potential land use impacts from license extension fail to consider the new evidence that such land use is likely to be indefinite and that the impacts of such indefinite land use at this site are substantial. Since the purpose of the GEIS is to accurately characterize the potential environmental impacts of the license extension it is apparent that unless evidence of indefinite spent fuel storage at the site is allowed and unless the environmental impacts on land use of such indefinite storage are considered, the GEIS will not serve its function. This problem is also correctable, without

amending the GEIS, by including the additional analysis in the SEIS as we suggest above.

<u>Third Contention</u> (Safety)

The Application must be denied because the Applicant has failed to fully identify plant systems, structures and components that are non-safety-related systems, structures, and components in the security area whose failure could prevent satisfactory accomplishment of any of the functions of safety-related systems, structures and components in accordance with 10 C.F.R. §54.4(a)(2), such that the Commission cannot find that 10 C.F.R. §54.29(a) is met.

This contention asserts that security equipment meets the definition of 10 C.F.R. §54.4.(a)(2) and that it should be demonstrated that the effects of aging on the functionality of this security equipment should be managed during the period of extended operation, just as it must be for all other equipment meeting the definition of 10 C.F.R. §54.4.(a)(2). Applicant and NRC Staff oppose admission of this contention on the grounds that it is outside the scope of the proceeding. The Staff also claims the contention is not material to the findings the NRC must make to support the action that is involved in the proceeding, does not set forth a specific factual or legal basis, as required, and does not demonstrate the existence of a genuine dispute on a material issue of law or fact.

1. Security equipment is not different than any other equipment meeting the definition of 10 C.F.R. §54.4.(a)(2).

Security equipment is non-safety equipment whose failure could compromise the functioning of safety equipment. *See* DPS Petition at 32.-3 Many non-safety systems, structures and components whose failure could prevent satisfactory accomplishment of safety related functions are screened out through the provisions of 10 C.F.R. §54.21(a)(1) as having moving

parts or with a change in configuration or properties, or are subject to replacement based on a qualified life or specified time period. This is also true of security equipment such as intrusion alarms, emergency alarms, communications equipment, and various interdiction weapons. Other security equipment, such as physical barriers and structures, would not be screened out by 10 C.F.R. §54.21(a)(1). Examples of such physical barriers and structures, which are visible upon entry to the plant complex, are concrete vehicle barriers and bullet resistant enclosures ("guard towers")[19]. Failure of a vehicle barrier through age degradation could allow entry of radiological saboteurs that could subsequently prevent satisfactory accomplishment of safety related functions. Failure of a bullet resistant enclosure through age degradation could admit radiological saboteurs whose actions could subsequently prevent satisfactory accomplishment of safety related functions. There is no reason that the age management provisions of 10 C.F.R. §54.21 (a) should not be applied to security equipment just as it is to other 10 C.F.R. §54.4.(a)(2) equipment.

Applicant's attempt at dismissing security equipment as not *directly* preventing satisfactory accomplishment of safety related functions (and creating a novel and unheard of standard, *fairly direct effect*) is entirely unpersuasive. Entergy Answer at 26-28. The age-degradation failure of a bullet resistant enclosure, vehicle barrier, or other item of security equipment could admit radiological saboteurs whose intent would be to prevent satisfactory

[19] Similar to footnote 6 of the DPS Petition, at 33, DPS is using vehicle barriers and bullet resistant enclosures as "non-Safeguards Information" examples of security equipment. Vehicle barriers and bullet resistant enclosures are visible and obvious to visitors to the station. DPS has not identified other specific systems, structures and components required by 10 C.F.R. Part 73 in order to avoid a Nuclear Safeguards Information designation. DPS continues to reserve its rights, under a rebuttal of lack of specificity on this contention, to file a list of systems, structures and components required by 10 C.F.R. Part 73 that require aging management review under 10 C.F.R. §54.21.

accomplishment of safety related functions. In San Luis Obispo Mothers for Peace v. NRC, No. 03-74628, slip op. (9th Cir. June 2, 2006) ("Mothers for Peace"), the Court held that NRC could not consider the possibility of an attack by radiological saboteurs as *remote and speculative* (in the case at hand, for the NEPA evaluation). Applicant seeks to discount the impact of this decision relying primarily on *Limerick Ecology Action v. NRC*, 869 F.2d 719, 741-44 (3d Cir. 1989) and *Duke Energy Corp.* (McGuire Nuclear Station, Units 1 and 2), CLI-02-26, 56 N.R.C. 358, 363 (2002). The former has little relevance, having been written prior to September 11, 2001, when the terrorist attacks in the United States became far less speculative and the urgency of evaluating them became far more important. The latter is inapplicable to Contention 3. DPS does not challenge the security measures taken which is the thrust of the *McGuire* decision. The Contention challenges Applicant's refusal to provide the type of long term maintenance and age management that is to be applied to all other equipment whose failure could impact the performance of safety-related equipment. In addition, to the extent the GEIS or any other regulations, all of which were written prior to 9/11, purport to excuse security equipment from age management, there is new and significant information that such actions were imprudent and need to be re-evaluated.

Barriers credited in the security plan are not different in function than fire barriers. Both are passive components. Both have design bases to prevent an occurrence for a time period - one due to fire, and the other due to radiological saboteur intrusion. Fire barriers are identified in the License Renewal Application, Sections 2.1.2.2.1, 2.1.2.4.2, 2.3.3.8; throughout the Tables of Section 2.4; Table 3.3.1, and Table 3.3.5. The age management program for fire barriers is described in Section B.1.12 of Appendix B. The same type of review and age management is

necessary for security systems, structures and components whose failure could prevent satisfactory accomplishment of safety related functions.

2. Security equipment should not be considered outside the scope of this proceeding

Security equipment is not excluded from consideration by any regulation. Rather, the link of applicability is stated in the DPS Petition, at 32:

> Plant systems, structures, and components within the scoping criteria of 10 C.F.R. §54.4 are not limited to systems, structures, and components required in accordance with 10 C.F.R. Part 50. Within the definition of current licensing basis in 10 C.F.R. §54.3, numerous Parts of 10 C.F.R. are identified, including 10 C.F.R. Part 73.

NRC Staff and Applicant rely only on the *statement of consideration* from 1991, now dated and stale as a result of September 11, 2001 terrorist attack and the Mothers for Peace decision. Both NRC Staff and Applicant quote the following:

> The requirements of 10 CFR part 73, notably the testing and maintenance requirements of 10 CFR 73.55(g), include provisions for keeping up the performance of security equipment against impairment due to age-related degradation or other causes. Once a licensee establishes an acceptable physical protection system, changes that would decrease the effectiveness of the system cannot be made without filing an application for license amendment in accordance with 10 CFR 50.54(p)(1).
>
> Application for a renewed license will not affect the standards for physical protection required by the NRC. The level of protection will be maintained during the renewal term in the same manner as during the original license term, since these requirements remain in effect during the renewal term by the language of § 54.35. The requirements of 10 CFR part 73 will continue to be reviewed and changed to incorporate new information, as necessary. The NRC will continue to ensure compliance of all licensees, whether operating under an original license or a renewed one, through ongoing inspections and reviews.

Final Rule, "Nuclear Power Plant License Renewal," 56 Fed. Reg. 64,943, 64,967 (Dec. 13, 1991) (1991 Final Rule). This logic emanates from the implicit regulatory notion, prevalent before September 11, 2001, that attack by radiological saboteurs is remote and speculative[20]. Therefore, the same detailed attention to age management was not given to security equipment as it was to other non-safety related equipment whose failure could prevent satisfactory accomplishment of safety related functions. Security equipment was primarily thought of as active equipment, such as intrusion alarms, emergency alarms, communications equipment, and various interdiction weapons, whose function would be demonstrated by the maintenance requirements of 10 C.F.R. §73.55(g).

Under closer scrutiny necessitated following the September 11, 2001 terrorist attack, it is clear that 10 C.F.R. §73.55(g) does not invoke the age management provisions on a level comparable to 10 C.F.R. §54.21 for security equipment such as vehicle barriers, bullet resistant enclosures, or other similar equipment. 10 C.F.R. §73.55(g)(1) requires only that " All alarms, communication equipment, physical barriers, and other security related devices or equipment shall be maintained in operable condition." No guidance is given for how the determination of operability is to be made for such equipment as vehicle barriers, bullet resistant enclosures, and other similar equipment. The requirements of 10 C.F.R. §54.21 are more detailed. Under 10 C.F.R. §54.21(a)(3), for vehicle barriers, bullet resistant enclosures, and other similar equipment, the Applicant would have to "demonstrate that the effects of aging will be adequately managed so that the intended function(s) will be maintained consistent with the CLB for the period of

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[20] The validity of this statement is proven by NRC's attempt to continue to hold to the *remote and speculative* position in Mothers for Peace, a position that is refuted by the Court.

extended operation."

Applicant's testing and maintenance program for security equipment in accordance with 10 C.F.R. §73.55(g) was established long before consideration of age degradation of vehicle barriers, bullet resistant enclosures, and other similar equipment were issues. There is no statement that Applicant's testing and maintenance program in accordance with 10 C.F.R. §73.55(g) includes provisions that demonstrate, for vehicle barriers, bullet resistant enclosures, and all other similar equipment, that the effects of aging will be adequately managed so that the intended function(s) will be maintained consistent with the CLB for the period of extended operation[21] [22].

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[21] Applicant had the opportunity to provide such statement by affidavit of a credible expert in its *Answer*, but did not.

[22] In addition, Applicant argues at 28 of its *Answer* from the *statement of consideration* for the maintenance rule, that "security has been deleted from 10 CFR 50.65 [i.e., the maintenance rule] as it is adequately addressed in § 73.46(g) and § 73.55(g)." This argument, intended to show that security systems, structures and components (SSCs) should not be considered under 10 C.F.R. §54.4(a)(2), instead proves the reverse, and confirms our argument at this point.

Maintenance of non-safety related SSC's whose failure could prevent satisfactory accomplishment of safety related functions, which are not security SSCs, is performed under 10 C.F.R. §50.65, the maintenance rule. The basic requirement of the maintenance rule is in 10 C.F.R. §50.65(a)(1), that these SSCs "*are capable of fulfilling their intended functions.*" Emphasis added.

Maintenance of security SSCs is performed under 10 C.F.R. §73.55 (g). The basic requirement of the security testing and maintenance requirement is that security SSCs "*shall be maintained in an operable condition.*" Emphasis added.

Reading of the two requirements shows they are parallel - essentially the same. Yet the non-safety SSCs under the maintenance rule are included for license renewal consideration under 10 C.F.R §54.4(a)(2). Therefore it makes no sense in logic to exclude security SSCs, as the Staff and Applicant quote for the 1991 *statement of consideration* for license renewal, when the testing and maintenance requirements are essentially identical for the SSCs that are included.

The explanation for this suspension of logic lies in the implicit underlying notion in the *statement of consideration* that security challenges by radiological saboteurs is remote and speculative. This notion is shown to be changed by the September 11, 2001 attacks and by

Staff includes an argument that DPS misreads 10 C.F.R. §54.21. Staff Answer at 20-21. Staff states that "not all SSCs within the scope of Section 54.4 are subject to management review." Then Staff quotes the McGuire and Catawba license renewal proceeding, that security SSCs are not subject to the physical aging processes at issue in license renewal. 56 NRC at 364. This statement and logic is simply not correct with regard to DPS Contention 3. For example, the concrete vehicle barriers have a design basis to prevent vehicle intrusion. As shown in Section 3.5 of the License Renewal Applicaiton (LRA), loss of material, scaling, cracking and spalling, are physical aging processes of concrete at issue in license renewal. Loss of material, scaling, cracking and spalling, could occur in a manner such that concrete vehicle barriers no longer meet their design basis for vehicle prevention.

Similarly, bullet resistant enclosures have a design basis to resist bullets. The bullet resistant material needs to be evaluated in a manner similar to the other materials age evaluations in the LRA, to prove the such material does not lose its bullet-resistance during the period of license renewal, or that the bullet-resistant nature of the material is monitored in a manner to ensure it continues to meet its design basis or that newer and more dangerous bullets have not been developed. Finally, the structural steel support of bullet resistant enclosures, of necessity, has a design basis related to radiological saboteur intrusion. Aging effects on structural steel is an aging process at issue in Section 3.5 of the LRA. The structural steel supports of the bullet resistant enclosures needs to be evaluated to prove the such material does not degrade in a manner to no longer meet its design basis, or is monitored in a manner to ensure it continues to meet its design basis.

Mothers for Peace.

As stated earlier, vehicle barriers and bullet resistant enclosures are non-Safeguards Information examples of security systems, structures and components. All security systems, structures and components need to be reviewed thoroughly and methodically, as required by 10 C.F.R. §.§54.4 and 54.21.

Since:

> 1) there exists security structures, systems and components that are subject to the physical aging processes at issue in license renewal;
>
> 2) the maintenance program in accordance with 10 C.F.R. §73.55(g) was established long before aging management issues were a consideration;
>
> 3) there is no statement that the maintenance program in accordance with 10 C.F.R. §73.55(g) demonstrates that the effects of aging will be adequately managed so that the intended function(s) will be maintained consistent with the CLB for the period of extended operation for all applicable security systems, structures and components; and
>
> 4) the entire paradigm for understanding of the significance and impact of radiological saboteurs has been completely transformed by the September 11, 2001 terrorist attacks and Mothers for Peace;

the *statement of consideration* from 1991 should not be considered determinative for security equipment in this proceeding, and security equipment should not be considered outside the scope of the proceeding.

3. The issue raised by Contention 3 is material to the findings NRC must make to approve the license renewal

NRC Staff claims that DPS fails to demonstrate the issue raised by Contention 3 is material to the findings NRC must make to approve the license renewal. However, absent the statement itself, Staff makes no argument supporting that claim. NRC Staff ignores the contention which states, in part "that the Commission cannot find that 10 C.F.R. §54.29(a) is

met." Section 54.29 (a) requires a determination that there is reasonable assurance that the activities authorized by the renewed license will continue to be conducted in accordance with the CLB, and that any changes made to the plant's CLB in order to comply with this paragraph are in accord with the Act and the Commission's regulations. These matters include managing the effects of aging during the period of extended operation on the functionality of structures and components that have been identified to require review under § 54.21(a)(1). We have shown above that security equipment is within the definition of 10 C.F.R. §54.4.(a)(2), and there are examples of security equipment that meet the evaluation requirements of § 54.21(a)(1). Therefore, the issue raised by Contention 3 is material to the findings NRC must make to approve the license renewal.

4. A specific factual basis of Contention 3 is provided

The staff also claimed DPS did not set forth a necessary factual basis for Contention 3. Staff Answer at 21. The DPS Petition, at 32, included the following:

> 3. 10 C.F.R. Part 73 requires the Applicant to provide systems, structures and components for physical protection of plant and materials. Specifically, systems, structures and components are required under Sections:
>
> > 73.40 Physical protection: General requirements at fixed sites.
> >
> > 73.45 Performance capabilities for fixed site physical protection systems.
> >
> > 73.46 Fixed site physical protection systems, subsystems, components, and procedures.
> >
> > 73.51 Requirements for the physical protection of stored spent nuclear fuel and high-level radioactive waste.
> >
> > 73.55 Requirements for physical protection of licensed activities in nuclear power reactors against radiological sabotage.

The above identifies that security equipment exists. DPS expected that Staff would be able to

agree that applicable security equipment exists. Footnote 6 (DPS Petition at 33) reserved the right to supplement under claim of lack of specificity. The Staff suggests at note 20 (Answer at 21) that "DPS does not explain why it failed to submit the information under seal,". The complexity of a state filing Nuclear Safeguards Information made such a filing infeasible. The difficulty of such filing, as noted above, is underscored by the fact that the DPS attorneys appearing in this case are not (at this time) authorized to view Safeguards Information[23].

While maintaining that the quoted item above from the DPS Petition at 32 is sufficient factual identification of security equipment, DPS has provided addition specific factual identification in this *Reply* for other reasons. Therefore, contrary to Staff claims, a specific factual basis is provided for Contention 3.

CONCLUSION

For all the reasons stated here and provided in the initial Petition DPS urges the Board to admit the Contentions to resolve the genuine dispute that exists between it and the Applicant regarding the facts and opinions which are at issue.

Respectfully submitted,

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National Legal Scholars Law Firm

[23] Vermont treats Safeguards Information with great care. Authorization to view Safeguards Information under 10 C.F.R. §73.21(c)(iii) is given under only the strictest standard of need to know.

84 East Thetford Rd.
Lyme, NH 03768

Dated this 30th day of June, 2006 at Montpelier, Vermont.

UNITED STATES
NUCLEAR REGULATORY COMMISSION

In Re: Entergy Nuclear Vermont Yankee	**)**	
LLC and Entergy Nuclear	**)**	**Docket No. 50-271**
Operations, Inc.	**)**	**ASLBP No. 06-849-03-LR**
(Vermont Yankee Nuclear Pwer Station)	**)**	

DECLARATION OF WILLIAM K. SHERMAN

accompanying

Vermont Department of Public Service
Reply to Answers of Applicant and NRC Staff
to Notice of Intention to Participate
and Petition to Intervene

William K. Sherman states as follows under penalties of perjury.

Introduction

1. My name is William K. Sherman. I am employed by the Vermont Public Service Department. My title is Vermont State Nuclear Engineer. I have held this position since November of 1988. My duties include ongoing State regulatory oversight of the Vermont Yankee Nuclear Power Station ("Vermont Yankee"), as well as advising the Department and other state agencies on issues related to Vermont Yankee and nuclear power. My professional and educational experience was summarized in the resume attached the Declaration filed with the Notice and Intention to Participate and Petition to Intervene.

2. My responsibilities with the Department include monitoring for the state of Vermont both the political and technical developments associated with management and ultimate disposal of nuclear waste.

3. I am providing this Declaration in support of the Vermont Department of Public Service Reply to Answers of Applicant and NRC Staff to Notice of Intention to Participate and Petition to Intervene ("DPS Reply").

4. I am familiar with the license amendment application for a license extension of twenty years submitted by Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc.

5. I assisted in the preparation of the DPS Reply.

6. The facts provided in my declaration are true and correct to the best of my knowledge and belief, and the opinions expressed herein are based on my best professional judgment.

7. The Exhibits attached to the DPS Reply are true and correct copies of the documents represented.

Primary Containment Concrete

8. I have performed a sample heat transfer calculation of a section through the drywell to assess the temperature on the face of the concrete outside the steel drywell. For the calculation, I have used *Marks' Standard Handbook for Mechanical Engineers*, Eighth Edition, 1978, McGraw Hill, pp. 4-59 to 4-70 (Transmission of Heat by Conduction and Convection).

9. For the heat transfer calculation, I have also used Entergy's *License Renewal Application, Amendment No. 2*, dated May 15, 2006 (Vermont Reply Exhibit 1). This submittal identifies that, above the transition zone from spherical to cylindrical portions, the drywell is separated from reinforced concrete by a two-inch gap. The gap below this transition is filled with sand. In addition, the Amendment refers to the nominal plate thickness of the drywell as 2.5 inches.

10. I used a representative cross section at El. 280 ft for the calculation. I assumed a steel plate of 2.5 inches, a sand-filled gap of 2 inches, and a concrete thickness of 6 feet.

11. I assumed the drywell temperature was 165°F, the maximum value from UFSAR Section 5.2.3.2, and a reactor building temperature of 100°F. I further assumed these areas were at these temperatures long enough such that the steel surface inside the drywell and the concrete surface temperature in the reactor building were at these respective temperatures.

12. I took thermal conductivities from the *Marks Handbook.* These values were, in units of btu/hr/ft^2/°F/ft: steel plate - 26.2, dry sand - 0.188, concrete - 1.05.

13. At equilibrium, the results of this temperature gradient are:

Temperature at steel surface in the drywell - 165°F

Temperature at the steel/sand interface - 164.9°F

Temperature at the inside concrete face - 156.2°F

In this calculation, approximately 8 inches of thickness of the concrete remains over 150°F.

14. This calculation confirms my statement from the Declaration for the Petition:

> "The concrete surface behind the steel shell will closely match the drywell ambient temperature."

This statement can be made by inspection - e.g., with a steel plate, small gap and approximately six foot thickness of concrete, the inside concrete surface temperature will be close to the ambient temperature on the face of the steel plate.

15. It is possible there are locations where the sand gap is less than two inches, or steel may touch concrete. In this case, the concrete temperatures would be higher. In other instances, concrete thickness is greater than 6 feet, which also would result in higher inside surface concrete temperatures.

16. If the area near the vicinity of the recirculating pump motors is maintained with a maximum of 135°F, and the average is 150°F, then the area away from the recirculating pump motors is at 165°F. Since the recirculating pump motors are toward the inside of the drywell near the reactor, the area away from recirculating pump motors are the outer walls of the drywell. It is the temperature on the outer walls that controls the heat transfer and gradient through the walls to the concrete outside the steel drywell.

17. Applicant suggests (Applicants Answer at 13) that it is inappropriate to use 165 °F for a general area temperature for concrete surface temperature determination. A portion of *Vermont Yankee Summary Report of Plant Environmental Conditions for Environmental Qualification Program,* Rev. 0, March 19, 1984, is provided as Vermont Reply Exhibit 2. This includes page 4, "Normal Operating Plant Environments," which includes drywell operating temperatures. This information was developed from actual thermocouple readings.

18. While this information is likely not current, at least it is representative of the thermal operation of the drywell that affected drywell concrete properties during a period of operation in 1980's. It is the temperature history that is relevant in the consideration of the attribute of *reduction of strength and modulus of the primary containment structure due to elevated temperature.* Therefore, this data from 1984 is specifically relevant. In addition, the temperature measurements were made when VY's maximum operation was 100% of thermal power. It now is allowed to operate at 120% of thermal power.

19. This information shows that, for the general area from El. 270 ft to El. 315 ft, an average temperature of 185 °F should be used.. This is an average temperature applicable for use for the general area, as opposed to the peak temperature, listed as 195 °F., which would be applicable for local area usage.

20. This information from the 1984 Vermont Reply Exhibit 2 demonstrates it is not incorrect to use 165 °F for the temperature next to the steel wall inside the drywell for determining the general area temperature for primary containment concrete outside the steel drywell between El. 270 ft. and El. 315 ft.

<u>Land Use</u>

21. I have reviewed the GEIS and based on that review it is my conclusion that there is no substantive analysis or discussion of the environmental impact associated with the loss of land use due to the continued storage of the spent fuel at the reactor site following the shutdown of a reactor. In particular, the GEIS does not consider that at individual sites the continued presence of spent fuel at the reactor site 1) may substantially interfere with the use and development of valuable land both at the reactor site and adjacent to the site and 2) may require a considerable commitment of economic resources from local and state authorities to maintain adequate support for safety and security required to be maintained throughout the time spent fuel remains at the site.

22. While individual statements, quoted by the Applicant, were made in the Waste Confidence Decision about unexpected results at Yucca mountain, the possible need for a second repository, and a possible reconsideration of reprocessing, it is the sum and combination of each of these occurring together which constitutes new and significant information.

23. There is no question that the discovery of groundwater at disposal levels at Yucca Mountain has created a compete paradigm shift.

24. A primary reason Yucca Mountain was chosen was because it contained a unique geological formation that was thought to prevent groundwater intrusion.

25. The fact that groundwater has been recently discovered and the paradigm for design has

shifted is seen in the U.S Nuclear Waste Technical Review Board's (NWTRB's) *Report to the U.S. Congress and Secretary of Energy*, January 1, 2005 to February 28, 2006 ("NWTRB Report"). The body of the NWTRB Report is filled with details related to having to create a new design for the groundwater that has been discovered.

26. The change in national policy for waste disposal also constitutes new and significant standard. As stated in the petition, the Administration is embarked on a major new initiative labeled the Global Nuclear Energy Partnership (GNEP). As part of GNEP, the Administration proposes changing to a novel mode of reprocessing in which unused Uranium would be removed but Plutonium would remain in a form that does not promote proliferation of weapons-grade nuclear material. This novel mode of reprocessing is unproven.

27. Prominent political supporters of GNEP advocate retaining spent nuclear fuel in its present location while the nation embarks on a research program with an undefined schedule to try to find a reprocessing process that would meet these goals. They reason that, since spent fuel will not be disposed of, but rather reprocessed, it should not be moved until after it can be reprocessed. And further, the disposal plan at the repository would have to undergo a major modification to accept reprocessing waste forms instead of spent fuel. The result is that all spent fuel disposal plans would be on hold while it is determined if (not when) a reprocessing method could be developed.

28. The Commission certainly did not envision in the Waste Confidence Decision a turn to a completely novel and unproven method of reprocessing with no set schedule and disposal plans on hold.

29. The Applicant gives short shrift (Answer at 21) to the changing political climate regarding spent nuclear fuel disposal. I believe most involved with the spent fuel disposal dilemma would say it is primarily a political problem. Part of the new and significant

information is the political landscape. We have an Administration, responsible for implementing spent fuel disposal, which is now promoting the novel GNEP. We have the most powerful nuclear advocate in the Senate, Sen. Domenici, also promoting GNEP and urging retention of spent fuel at their current locatoins. We have the most powerful Senator for the other party, Sen. Reid, as the primary opponent of Yucca Mountain development, also urging retention of spent fuel on their present sites indefinitely. This political landscape constitutes new and significant information which will have high impact on whether spent nuclear fuel will ever move, and the land use at Vermont Yankee.

30. With the changed political landscape, there is no basis to believe "Congress will provide the needed institutional support and funding, as it has for the first repository."

31. If it took from 1985 until 2025, a period of 40 years, to develop the first repository, there is no basis to believe that a second repository, if started immediately, could be developed within 34 years, given the past history and present political landscape.

32. It is not each single item mentioned about that constitutes new and significant information, but it is the sum of all of these items that results in a situation where spent nuclear fuel will remain at Vermont indefinitely, creating a MODERATE to LARGE evaluation associated with this use of land.

Security

33. Many non-safety systems, structures and components whose failure could prevent satisfactory accomplishment of safety related functions are screened out through the provisions of 10 C.F.R. §54.21(a)(1) as having moving parts or with a change in configuration or properties, or are subject to replacement based on a qualified life or specified time period. This is also true of security equipment such as intrusion alarms, emergency alarms, communications equipment, and various interdiction weapons.

34. Other security equipment, such as physical barriers and structures, would not be screened out by 10 C.F.R. §54.21(a)(1). Examples of such physical barriers and structures, which are visible upon entry to the plant complex, are concrete vehicle barriers and bullet resistant enclosures ("guard towers").

35. Failure of a vehicle barrier through age degradation could allow entry of the vehicle of radiological saboteurs that could subsequently prevent satisfactory accomplishment of safety related functions.

36. Failure of a bullet resistant enclosure through age degradation could admit radiological saboteurs whose actions could subsequently prevent satisfactory accomplishment of safety related functions.

37. There is no reason that the age management provisions of 10 C.F.R. §54.21 (a) should not be applied to security equipment just as it is to other 10 C.F.R. §54.4.(a)(2) equipment.

38. The age-degradation failure of a bullet resistant enclosure, vehicle barrier, or other item of security equipment could admit radiological saboteurs whose intent would be to prevent satisfactory accomplishment of safety related functions.

39. Barriers credited in the security plan are not different in function than fire barriers. Both are passive components. Both have design basis to prevent an occurrence for a time period - one due to fire, and the other due to radiological saboteur intrusion. The same type of review and age management is necessary for security systems, structures and components whose failure could prevent satisfactory accomplishment of safety related functions.

40. Before September 11, 2001 attack by radiological saboteurs was considered remote and

speculative. The same detailed attention to age management was not given to security equipment as it was to other non-safety related equipment whose failure could prevent satisfactory accomplishment of safety related functions. Security equipment was primarily thought of as active equipment, such as intrusion alarms, emergency alarms, communications equipment, and various interdiction weapons, whose function would be demonstrated by the maintenance requirements of 10 C.F.R. §73.55(g).

41. Applicant's testing and maintenance program for security equipment in accordance with 10 C.F.R. §73.55(g) was established long before consideration of age degradation of vehicle barriers, bullet resistant enclosures, and other similar equipment were issues.

42. Maintenance of non-safety related SSC's whose failure could prevent satisfactory accomplishment of safety related functions, which are not security SSCs, is performed under 10 C.F.R. §50.65, the maintenance rule. The basic requirement of the maintenance rule is in 10 C.F.R. §50.65(a)(1), that these SSCs "*are capable of fulfilling their intended functions*." Maintenance of security SSCs is performed under 10 C.F.R. §73.55 (g). The basic requirement of the security testing and maintenance requirement is that security SSCs "*shall be maintained in an operable condition*." These two requirements are essentially the same. .

43. It is not logical to exclude security SSCs, based on maintenance and testing requirements, when the testing and maintenance requirements are essentially identical for the SSCs that are included. The explanation for this suspension of logic lies in the now *passe* notion that security challenges by radiological saboteurs is remote and speculative.

44. Concrete vehicle barriers have a design basis to prevent vehicle intrusion. As shown in Section 3.5 of the License Renewal Applicaiton (LRA), loss of material, scaling, cracking and spalling, are physical aging processes of concrete at issue in license renewal. Loss of material, scaling, cracking and spalling, could occur in a manner such that concrete

vehicle barriers no longer meet their design basis for vehicle prevention.

45. Bullet resistant enclosures have a design basis to resist bullets. The bullet resistant material needs to be evaluated in a manner similar to the other materials age evaluations in the LRA, to prove the such material does not lose its bullet-resistance during the period of license renewal, or that the bullet-resistant nature of the material is monitored in a manner to ensure it continues to meet its design basis.

46. Structural steel support of bullet resistant enclosures, of necessity, has a design basis related to radiological saboteur intrusion. Aging effects on structural steel is an aging process at issue in Section 3.5 of the LRA. The structural steel supports of the bullet resistant enclosures needs to be evaluated to prove the such material does not degrade in a manner to no longer meet its design basis, or is monitored in a manner to ensure it continues to meet its design basis.

47. The requirements for a Vermont to make a Nuclear Safeguards Information filing are cumbersome, even unduly so, far more than a simple non-disclosure agreement. The attorneys entered on this case for the DPS are not (at this time) authorized to view Safeguards Information, as I am. Vermont treats Safeguards Information with great care. Authorization to view Safeguards Information under 10 C.F.R. §73.21(c)(iii) is given under only the strictest standard of need to know.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on June 30, 2006.

William K. Sherman

William K. Sherman
State Nuclear Engineer

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May 15, 2006

BVY 06-043

ATTN: Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Reference: 1. Letter, Entergy to USNRC, "Vermont Yankee Nuclear Power Station, License No. DPR-28, License Renewal Application," BVY 06-009, dated January 25, 2006

Subject: **Vermont Yankee Nuclear Power Station**
License No. DPR-28 (Docket No. 50-271)
License Renewal Application, Amendment No. 2

On January 25, 2006, Entergy Nuclear Operations, Inc. and Entergy Nuclear Vermont Yankee, LLC (Entergy) submitted the license renewal application for the Vermont Yankee Nuclear Power Station (VYNPS) as indicated by Reference 1. Based on recent discussions between industry and NRC staff, Entergy is providing Attachment 1 to provide additional information concerning the drywell shell.

Just prior to the submittal of this letter, proposed license renewal interim staff guidance was published in the Federal Register (May 9, 2006). The NRC proposed guidance, "LR-ISG-01: Plant-Specific Aging Management Program for Inaccessible Areas of Boiling Water Reactor Mark I Steel Containment Drywell Shell," was issued for public comment. The proposed guidance is expected to be finalized by NRC staff after the comment period.

This letter contains no regulatory commitments.

Should you have any questions concerning this letter, please contact Mr. Jim DeVincentis at (802) 258-4236.

I declare under penalty of perjury that the foregoing is true and correct. Executed on May 15, 2006.

Sincerely,

Ted A. Sullivan
Site Vice President
Vermont Yankee Nuclear Power Station

Attachment (1)

cc: (on next page)

Docket 50-271-LR
ALSBP No. 06-849-03-LR
Exhibit Vermont Reply-1
10 Pages

A117

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Docket 50-271
BVY 06-043

Attachment 1

Vermont Yankee Nuclear Power Station

License Renewal Application – Amendment No. 2

Drywell Shell Information

Purpose

For license renewal, the NRC evaluates the potential for corrosion of the Mark I steel containment drywell shell. This issue previously was the subject of generic NRC communications in the 1980s. Specifically, Generic Letter (GL) 87-05 addressed potential degradation of Mark I drywells due to corrosion. This document provides additional information on the Vermont Yankee Nuclear Power Station (VYNPS) drywell shell relative to recent industry experience in this area.

Background

In 1980, the Oyster Creek Station observed water coming from lines that drain water from the annulus region between the drywell wall and the surrounding concrete and the sand cushion region. The water source was initially identified in 1983 as coming from the Drywell-Refueling Cavity bellows drain line gasket. After performing ultrasonic thickness measurements in 1986, Oyster Creek Station reported that corrosion and material loss had occurred to the Drywell Shell in the area of the sand-cushion. This led to the NRC's issuance of Information Notice 86-99 (Degradation of Steel Containments), Generic Letter 87-05 (Request for Additional Information Assessment of Licensee Measures to Mitigate and/or Identify Potential Degradation of Mark I Drywells), and Information Notice 86-99 Supplement 1.

The purpose of GL 87-05 was "...to initiate the collection of information of the licensee's current and proposed action to assure the degradation of the Drywell. Shell plates adjacent to the sand-cushion has not occurred and to determine if augmented inspections above and beyond those planned by the licensee's are necessary."

In 1995, subsequent to the GL responses, the staff approved the use of ASME Section XI, Subsection IWE (Requirements for Class MC and Metallic Liners of Class CC Components of Light-Water Cooled Plants) which exempts, in accordance with Subparagraph IWE-1220(b), "embedded or inaccessible portions of containment vessels, parts, and appurtenances that met the requirements of the original Construction Code..." However, Paragraph IWE-1240 establishes criteria for determining the need for augmented examinations.

VYNPS Primary Containment Design

At VYNPS, the primary containment includes the drywell, the suppression chamber, and the drywell to suppression chamber vent headers. The drywell is an inverted light bulb-shaped carbon steel primary containment structure enclosed in reinforced concrete founded on bedrock. Above the transition zone between the spherical and cylindrical portions, the drywell is separated from the reinforced concrete by a two-inch gap. This gap allows for drywell expansion.

Drywell Shell Exterior

A sand-filled cavity encircles the drywell to cushion the concrete to free standing steel transition. This sand cushion is equipped with drains to remove any water that might enter the sand and cause accelerated corrosion of the drywell shell. The sand cushion area is drained to protect the exterior surface of the drywell shell at the sand cushion interface from water that might enter the air gap.

During construction, the exterior surface of the drywell shell was coated with an inorganic zinc primer and a protective top coat. The coating is intact in areas that have been examined.

A pliable bellows assembly between the drywell shell and the refueling cavity (area 'A' on the enclosed general arrangement drawing) separates the filled refueling cavity from the exterior surface of the drywell shell during refueling operations. The assembly utilizes a fully welded stainless steel to carbon steel design, providing a channel to collect any potential leakage from the bellows. Leakage, if any, through the bellows assembly is directed to a drain system equipped with an alarm for notification of operators. While the refueling cavity is filled, plant operators examine areas around the drywell shell exterior to determine if leakage is occurring.

An additional source of water that could impact the drywell shell exterior is leakage from the spent fuel storage pool and dryer-separator pit liner welds. Channels behind the welds direct leakage, if any, to funnels. These funnels are routinely inspected by plant operators to determine if leakage exists from the spent fuel storage pool, the dryer-separator pit, or the refueling cavity drains. The majority of the drywell shell exterior surface is inaccessible for examination.

Drywell Shell Interior

The majority of upper portion of the drywell shell interior surfaces are accessible for inspection. The lower portion of the drywell is not accessible where it is covered by the concrete drywell floor which provides structural support for the reactor pedestal and other equipment.

The VYNPS primary containment system is inerted with nitrogen gas during normal power operations so that oxygen levels are maintained at less than 4%. Inerting with nitrogen provides an atmosphere that is not conducive to corrosion of containment interior surfaces.

Operating Experience and Actions Taken to Prevent Drywell Corrosion

VYNPS responded to GL 87-05 on May 8, 1987 indicating no evidence of degradation to the drywell was noted. Further, VYNPS committed to ensure continued drywell integrity via IWE inspection and inspections (including internals) of the eight 1" sand cushion drain lines for integrity and freedom from obstruction.

VYNPS reported on the refuel cavity design, explaining that the design is a fully welded stainless steel/carbon steel construction (vice Oyster Creek design) with a backup barrier channel that utilizes a seal (i.e., bellows) rupture drain with an alarm system for notifying operators in the event of any bellows or drain line connection leakage.

In 1991, during normal operations, leakage from a main steam line drain valve was condensing on and traveling along the primary containment atmosphere control piping to the drywell shell exterior. The typical penetration design slopes piping away from the drywell however, this atypical penetration is sloped towards the drywell. To ensure drywell shell integrity, the exterior drywell shell in the area of the sand cushion and the sand cushion itself (area 'B' on the enclosed general arrangement drawing) were examined by boroscope and the sand cushion drains were verified functional. No corrosion was found on the drywell shell and the sand cushion was found dry, compacted, and with adequate ventilation to assure the sand would remain dry. Spray shields were installed on piping penetrations that sloped towards the drywell shell.

A periodic surveillance (approximately every 10 years) was established to examine the drywell shell sand cushion drain lines for integrity and freedom from obstructions.

In 1992, the drywell interior, in the area of the sand-cushion was examined. The examination identified a missing section of the moisture barrier at the concrete floor to drywell shell interface joint (area 'C' on the enclosed general arrangement drawing). No evidence of corrosion of the interior drywell shell surface was observed. In 1999, during the implementation of the ASME Section XI IWE Program, corrosion was identified on the interior surface of the drywell shell in the area of the missing moisture barrier. The maximum pit depth was 1/16". The nominal plate thickness of the drywell shell in that area is 2.5".

In 2001 a replacement moisture barrier was installed. Prior to installation, the drywell shell interior and the concrete floor were stripped of all coatings and sealant for approximately a six inch band either side of the intersecting joint. The corrosion was removed. The drywell shell was then examined by VT-3, VT-1, and UT measurement processes. Observations and measurements met acceptance criteria. The replacement moisture barrier was installed. The moisture barrier was subsequently examined in 2002, 2004 and 2005. The examination evaluated the adherence of the drywell shell coating, no evidence of corrosion, elastomer to shell and concrete interface, and hardening of the elastomer.

Ongoing actions to Prevent Drywell Degradation

During approximately 95% of a fuel cycle, the VY primary containment system atmosphere is inerted with nitrogen. During this period, the atmosphere oxygen concentration is maintained less than 4%. The moisture content is reduced by a dehumidification system. Condensate from the dehumidification system is routed to dedicated drain lines and collected in sumps. The result is that the drywell interior is dry and oxygen-free at a relatively constant temperature that does not promote corrosion.

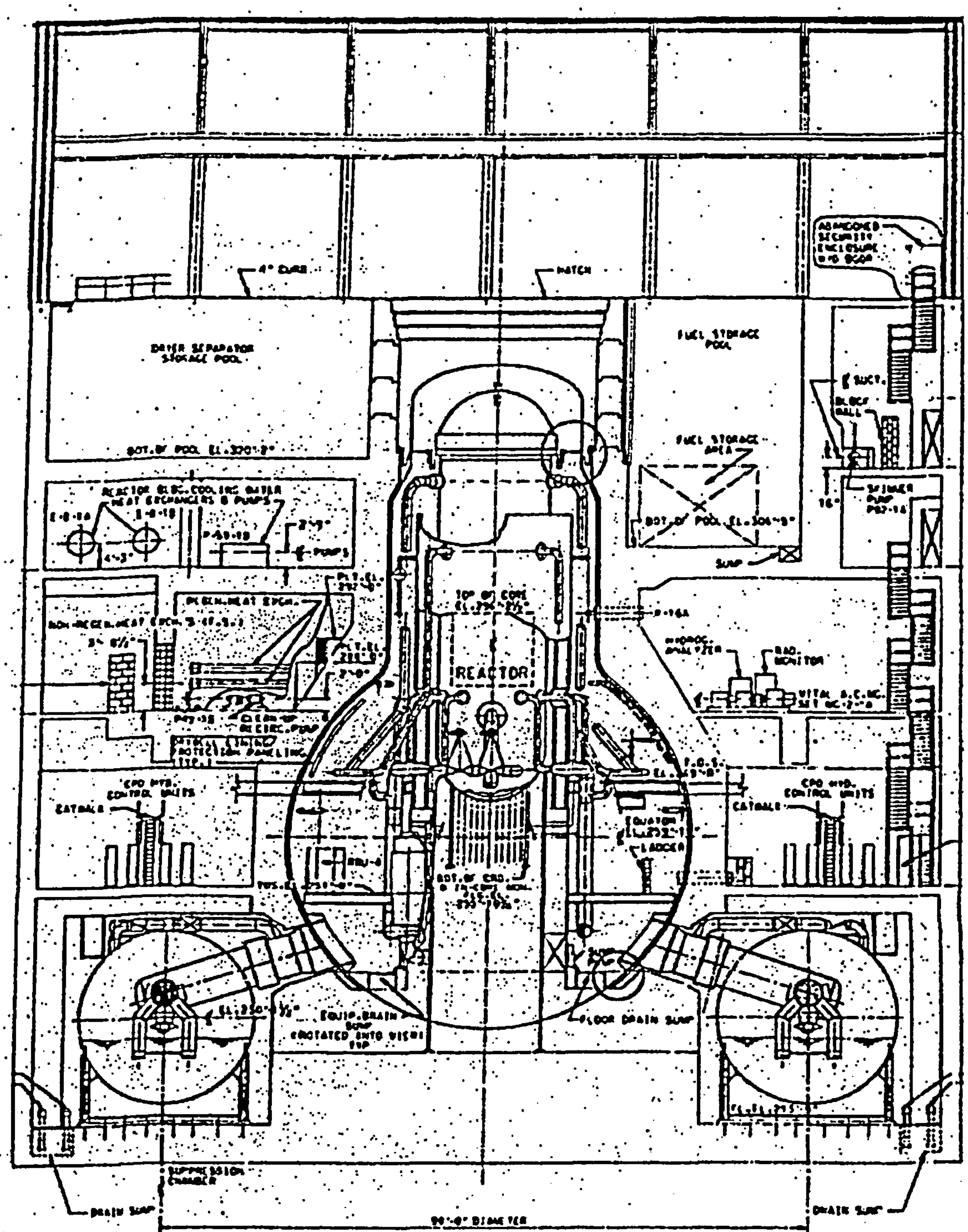
The suppression chamber exterior and interior surfaces, the majority of the vent header exterior and interior surfaces, the majority of the drywell shell interior surfaces, the drywell hemi-spherical head exterior and interior surfaces, and some penetrations in the cylindrical and spherical portions of the structure are accessible for examination. The structures are examined in accordance with ASME Section XI – 1998 Edition with 2000 Addenda, Subsection IWE, Requirements for Class MC and Metallic Liners of Class CC Components of Light-Water Cooled Plants. The accessible portions of the drywell shell interior surfaces are examined in accordance with the ASME code, three times during each ISI ten-year interval. As of May 2006, no surface areas are subject to the requirements of Paragraph IWE-1240, "Surface Areas Requiring Augmented Examination."

The moisture barrier is examined at least once every period, in accordance with ASME Section XI inservice inspection requirements.

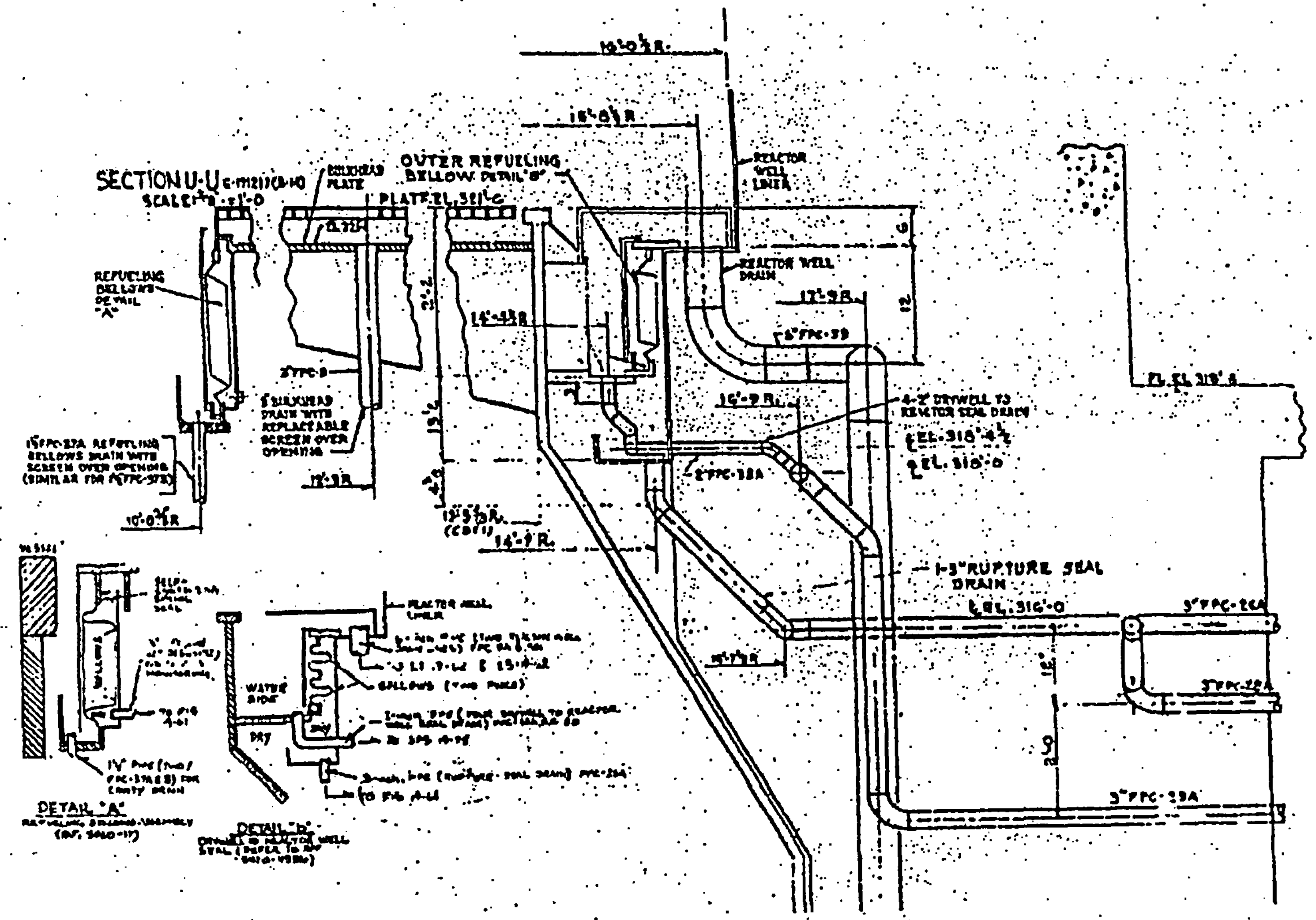
Approximately once every 10 years, the drywell shell sand cushion drain lines are examined to verify integrity and freedom from obstructions.

Conclusion

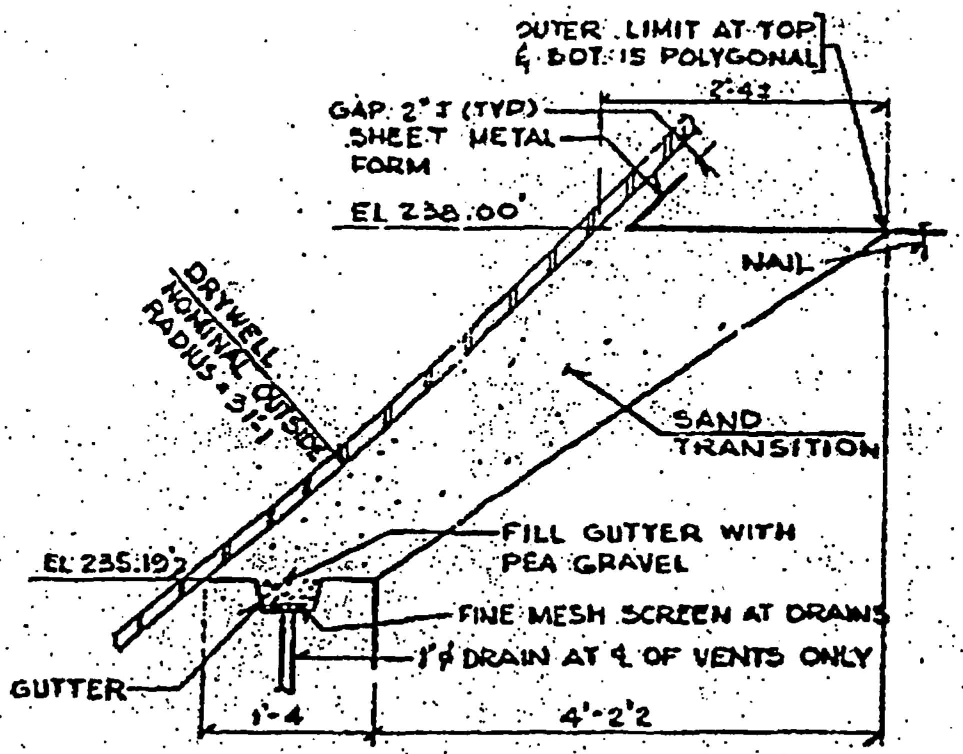
VYNPS has effectively addressed the issue of drywell shell corrosion through actions taken in response to GL 87-05 as well as additional actions subsequent to the response to GL 87-05. UT examinations to determine the drywell wall thickness at the sand cushion region indicated no detectable loss of material and hence no discernable corrosion rate. Based on this corrosion rate, no discernable loss of drywell shell thickness is projected through the period of extended operation. The above described ongoing actions to prevent drywell shell degradation provide continuing reasonable assurance of satisfactory drywell shell condition through the period of extended operation.



General Arrangement – Reactor Building
from G-191150 [Sect A-A]



VY Refueling Bellows and Drywell to Reactor Well Seal
from G-191277



DETAIL OF SAND TRANSITION
AT ℄ OF SEGMENT

VERMONT YANKEE SUMMARY REPORT

OF

PLANT ENVIRONMENTAL CONDITIONS

FOR ENVIRONMENTAL QUALIFICATION PROGRAM

Revision 0

March 19, 1984

D. E. Yasi

Docket 50-271-LR
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Exhibit Vermont Reply-2
3 Pages

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3.0 NORMAL OPERATING PLANT ENVIRONMENTS

Although well-documented measurements of environmental data throughout the plant have not been recorded, we believe the following data adequately represents the normal range of plant environmental conditions for use in evaluating normal aging effects of equipment. However, localized conditions due to high temperature piping and equipment could be considerably different from average conditions.

3.1 Temperature, Pressure, and Humidity

REACTOR BUILDING - OCCUPIED AREAS ONLY

(Excluding Primary Containment, Steam Tunnel, RCIC Turbine Room, HPCI Turbine Room)

	JAN	FEB	MAR	APR	MAY	JUN	JUL	AUG	SEP	OCT	NOV	DEC
Avg. Temp. °F	70	70	70	75	80	85	95	95	85	80	70	70
Avg. Humidity %	40	40	45	60	65	70	70	75	75	60	45	40

Peak Temperature: 104°F

Pressure: Ambient

DRYWELL (Operating and Hot Standby Modes)

	(Below El. 270') JAN - DEC	(El. 270' to El. 315') JAN - DEC	(Above El. 315') JAN - DEC
Avg. Temp. °F	150	185	270
Avg. Humidity %	<40	<40	<40
Peak Temperature: °F	160	195	280
Pressure: PSIG	2	2	2

Note: The average temperatures listed are based upon the hottest recorded location within each zone during plant operation. These values should be expected during 90% of plant life. For the remaining 10% of time, when the reactor is shutdown, an average temperature of 100°F will be experienced throughout the drywell.

When necessary, the local temperature near a particular component can be documented and utilized for aging calculations in lieu of the above temperatures. Average temperatures at many thermocouple locations in the drywell are documented in Appendix A.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
ENTERGY NUCLEAR VERMONT	)	Docket No. 50-271-LR
YANKEE LLC AND ENTERGY NUCLEAR	)	ASLBP No. 06-849-03-LR
OPERATIONS, INC.	)	
(Vermont Yankee Nuclear Power Station)	)	

CERTIFICATE OF SERVICE

I hereby certify that copies of the Corrected Copy Submitted on 7/6/06 of the Vermont Department of Public Service Reply to Answers of Applicant and NRC Staff to Notice of Intention to Participate and Petition to Intervene were served on the persons listed below by deposit in the U.S. Mail, first class, postage prepaid, on the 7th day of July, 2006, and by electronic mail and where indicated by an asterisk on this 6th day of July 2006.

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Respectfully submitted,

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