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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING BOARD

BEFORE THE COMMISSION

DOCKETED
USNRC

June 11, 2008 (4:05pm)

OFFICE OF SECRETARY
RULEMAKINGS AND
ADJUDICATIONS STAFF

In the Matter of	)	
	)	Docket No. 50-0219-LR
AMERGEN ENERGY COMPANY, LLC	)	
	)	
(License Renewal for the Oyster Creek	)	
Nuclear Generating Station)	)	
	)	

CITIZENS' RESPONSE TO COMMISSION ORDER DATED MAY 28, 2008

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June 11, 2008

TEMPLATE = SECY-035

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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION
OFFICE OF THE SECRETARY

BEFORE THE COMMISSION

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This response to the Commission's May 28, 2008 Order, is filed on behalf of Nuclear Information and Resource Service, Jersey Shore Nuclear Watch, Grandmothers, Mothers and More for Energy Safety, New Jersey Public Interest Research Group, New Jersey Sierra Club, and New Jersey Environmental Federation (collectively "Citizens"). The Commission has asked first whether AmerGen Energy Co. LLC ("AmerGen") has committed to do an analysis of the factor of safety that matches or bounds the sensitivity analyses that Judge Baratta would impose and second whether additional analysis is needed. The short answer is no and yes.

At present AmerGen has made a vague commitment to perform a three dimensional finite element structural analysis of the drywell shell using modern methods and the current drywell shell thickness. The analysis will include some sensitivity studies to determine how uncertainties in the size of severely corroded areas affect the margins. In contrast, Judge Baratta would impose a requirement to carry out a series of sensitivity analyses, at least one of which would use all the measured thickness data in an extrapolation scheme, that could be similar to that employed by Citizens' expert Dr. Hausler, to determine drywell thicknesses. To date AmerGen has not committed to use all the data, use such an extrapolation scheme, or carry out "a series of sensitivity analyses."

Judge Baratta's additional requirements will lead to extremely valuable information about the compliance of the drywell with the Current Licensing Basis ("CLB") requirement that there be a factor of safety of 2.0 during refueling, which is the requirement that is most stringent in terms of drywell thickness. However, these requirements are likely to further illustrate that the uncertainty about the current state of the drywell makes it impossible to assure compliance with the CLB with any certainty without further measurements. AmerGen should therefore be required to explicitly estimate the full range of uncertainty in the predicted factors of safety using a series of sensitivity analyses. Then, if compliance with the CLB is not demonstrated with a high level of certainty,[1] AmerGen should be required to perform additional measurements of both the thickness and the shape of the drywell to reduce the uncertainty in the model. AmerGen should then be required to do a re-analysis using the additional thickness measurements and the actual shape of the drywell to attempt to establish compliance with the CLB at the Required Level of Certainty. Armed with these results, the Commission could then take an appropriate decision on whether there is reasonable assurance that the drywell shell at the Oyster Creek Nuclear Generating Station ("Oyster Creek") would meet the CLB on the first day of any extended period of operation and would continue to do so.

ARGUMENT

I. AmerGen's Commitments Are Vague And Inconsistent With Reasonable Assurance

AmerGen has made a vague commitment to perform a three dimensional finite element structural analysis of the drywell shell using modern methods and using the current drywell shell thickness. NRC Staff Ex. 1 at A-30 to A-31. The analysis will include some sensitivity studies to determine how

[1] Based on legal and general technical arguments, Citizens have stated that the required level of certainty is at least 95%. However, it is possible that a higher degree of certainty is required to remain consistent with assumptions made in the risk assessments regarding accidents at Oyster Creek, which is an issue that Citizens have not examined. Citizens note that in the context of initial licensing 10 C.F.R. § 50.46 requires explicit estimates of uncertainty in calculated results and the analyses must show "a high level of probability that the [acceptance] criteria would not be exceeded." Furthermore, when initially licensed, the predictions of the factor of safety were much more certain than they are now, because the wall thicknesses were known to be at or very close to nominal design thicknesses. Moreover, compliance was assured by walls that were almost double the current thickness of the drywell in the areas that have corroded most. Because the Commission has not yet decided what level of certainty is required to establish reasonable assurance of compliance with the factor of safety requirements at Oyster Creek, Citizens will refer to the certainty required as the "Required Level of Certainty."

uncertainties in the size of thinned areas affect the margins. *Id.* at A-31. AmerGen committed to notify the NRC if "the analysis determines that the drywell shell does not meet required thickness values." *Id.* During this proceeding, the Atomic Safety and Licensing Board (the "Board") found that the CLB includes a requirement to meet the safety factor of 2.0 during refueling. LBP-07-17, 66 NRC 327 at n. 20. Thus, Citizens believe that AmerGen has committed to notify the NRC if the model predicts a factor of safety of less than 2.0 during refueling or fails to meet other similar requirements.

The notification aspect of this commitment is straightforwardly inadequate because the modeling should be required to affirmatively establish compliance with the CLB with reasonable assurance. As a matter of policy, uncertainty must be resolved against licensees where they control the level of certainty provided through decisions on scope and frequency of measurements. Thus, at minimum, AmerGen should be required to notify the NRC if the outcome of the modeling is indeterminate and fails to establish compliance with the factor of safety requirements of the CLB to the Required Level of Certainty.[2] Futhermore, because this modeling will serve as critical evidence concerning the resolution of the contention, AmerGen should be required to provide a copy of the analysis to Citizens irrespective of the outcome.

II. Judge Baratta Correctly Concluded That More Work Is Needed

As the Commission has correctly recognized, Judge Baratta believes that the licensee failed to "fully" show that "there is reasonable assurance that the factor of safety required by the regulations will be met throughout the period of extended operation . . . ." LBP-07-17, 66 NRC 327, 373 (Additional Statement of Judge Baratta at 1). This is because "to date . . . no analysis of the <u>actual</u> condition of the drywell has been done." *Id.* at 4 (emphasis in original). Therefore, "we do not know what the actual safety factor is." *Id.* Adding to the uncertainty caused by this lack of analysis is "a very limited knowledge of the actual thickness of the shell" because "there are large areas of the drywell that do not have any recent measurements or any measurements at all." *Id.* at 5. Because further corrosion of the drywell cannot be ruled out, "it is essential to have a conservative best estimate analysis of the drywell

[2] As discussed below, this modeling should be done prior to any decision on relicensing.

shell before entering the period of extended operation." *Id.* at 4. In addition, that analysis must take account of the uncertainty caused by the lack of knowledge regarding the thickness of the drywell. *Id.* at 5.

Judge Baratta recognized that AmerGen was going to use a three dimensional finite element model, which would have as inputs the measured thicknesses. *Id.* Based upon AmerGen's oral testimony he also concluded that the model will use the actual geometries of the drywell. *Id.*; *See also* Tr. at 659-60 (Gallagher).[3] He then stated "I would impose an *additional* requirement on the . . . applicant." Additional Statement of Judge Baratta at 6 (emphasis added). The additional requirement is that the applicant should perform a series of sensitivity analyses. *Id.* One of these analyses should include the use of an extrapolation method to determine the thickness between the measured locations. *Id.* This could be similar to the approach suggested by Citizens' expert and use contour plots generated from known thickness points measured from both the interior and the exterior. *Id.*

Plainly, Judge Baratta would not have suggested these additional requirements if he thought AmerGen had already committed to this analysis. In fact, AmerGen's commitments are much less detailed than the requirements spelled out by Judge Baratta. Furthermore, AmerGen documents show that AmerGen changed its mind on how to derive the thickness values for the three dimensional analysis, but has never proposed using all the measured results. *See* Citizens Ex. 65 (base the three dimensional analysis on the external measurements); Citizens Ex. 45 (use primarily the internal measurements). Moreover, AmerGen vigorously opposed the admission and then the use of Dr. Hausler's contour plots during this litigation. *E.g.* AmerGen Ex. C part 3 at 30-32.[4] Thus, Judge Baratta's additional

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[3] Contrary to the plain meaning of Mr. Gallagher's statement, information available to Citizens indicates that AmerGen intends to use the idealized geometry model it used in the past by modeling a geometrically idealized drywell and then applying a capacity reduction factor to take account of the real imperfect shape of the drywell. NRC Staff Ex. 3 at 5-4 to 5-5; Citizens Ex. 65 at 1.

[4] Part of the objection is that some of the external measurement points were selected to be at the visually identified minima and some may have been overground during surface preparation. Citizens recognize that the data set is less than ideal, but have analyzed these issues in detail and have found that these effects are insignificant. Citizens Proposed Finding of Fact and Conclusions of Law at 18-20. Thus, as Judge Baratta recognized, all the thickness data should be included in the further modeling by using contour plotting or a similar approach.

requirements not only go beyond what AmerGen has committed to do, they also go well beyond what AmerGen would do in the absence of any further regulatory requirements.

III. Existing Data Are Probably Insufficient To Provide Reasonable Assurance

All indications are that the drywell is, at best, marginally above the factor of safety requirements, but that there is a chance that it could already have violated the standards. For example, on August 17, 2007, Dr. Hartzman of the NRC Staff stated baldly that "Based on the currently available corrosion data of the sand bed region, the Staff estimates that the EFS [effective factor of safety] in the sand bed shell is 1.9." Affidavit of Mark Hartzman, dated August 17, 2007. However, the Staff then amended the pre-filed testimony to read: "Assuming that the corrosion is as extensive and severe as depicted by Dr. Hausler's contour plots in Citizens Exhibit 13, the Staff estimates that the EFS in the sand bed shell is 1.9." *See* NRC Staff Ex. C at A28. Subsequently on September 24, 2007, Dr. Hartzman testified orally that based on the average thickness of the shell, the current factor of safety is "probably about two, even greater than two." Tr. 453:12-16.

NRC Staff have shown no errors in Dr. Hausler's contour plots and Judge Baratta has recognized their value. On sur-rebuttal, Dr. Hausler showed that the plots he had presented previously did not show the full extent of corrosion because they were confined to the measured area as provided by AmerGen. Dr. Hausler then presented additional plots utilizing an additional extrapolation method showing even more extensive corrosion. Citizens' Ex. 61 at 14-17. Furthermore, Dr. Hausler showed that his plots were merely more refined versions of AmerGen's estimates regarding severely corroded areas. Citizens Ex. 61 at 4. Thus, if Dr. Hartzman had continued to rely on Dr. Hausler's plots or, had relied upon AmerGen's latest interpretation of the data, he would have predicted a factor of safety of less than 1.9. Furthermore, the NRC Staff repeatedly stated they had not reviewed AmerGen's latest analysis of the severely corroded areas in detail. NRC Staff Ex. B at A9 (page 13); Tr. 415:16-21; Tr. 420:4-10. Therefore, Dr. Hartzman's estimate that the factor of safety is greater than 2.0 based on the average thickness is undermined by his estimate that it is approximately 1.9, based upon more detailed data analysis.

Somewhat similarly, Dr. Mehta of GE testified that the factor of safety is probably "greater than two," but not much greater than two, but failed to state how he had interpreted the thickness measurements to reach this conclusion. Tr. 441:11-24. Finally, the best available analysis of the drywell shell was carried out by Sandia National Laboratories. Based on non-conservative assumptions, the Sandia Study concluded that the factor of safety was approximately 2.15 and the current buckling strength is approximately 44% lower than when it was built. Citizens' Proposed Conclusions of Fact and Law at 34-35. Sandia warned that its analysis was intended to focus on the relative reduction in design margin, rather than the absolute stress, and was insufficient for licensing purposes. NRC Staff Ex. 6 at 12. This was because many assumptions had to be made and thickness measurements were "limited to a few selected regions in the sandbed [region of the drywell shell]." *Id.* Judge Baratta also warned that the Sandia Study is "based on a very limited knowledge of the actual thickness of the drywell." Additional Statement of Judge Baratta at 3.

During the hearing, Judge Abramson also commented on the "unknown information" highlighted by Sandia and asked AmerGen if it was going to do "a lot of measuring." Tr. at 657:13-16. Surprisingly, even though AmerGen agrees that there are "an insufficient number of UT measurements to evaluate a representative average thickness over each area," *e.g.* AmerGen Ex. C at Part 3 A.38, AmerGen responded that it is going to base the required three dimensional analysis on the existing measurements. Tr. at 657:17:18. As Dr. Hausler has repeatedly pointed out, while it is possible to interpolate and extrapolate to generate thickness values for unmeasured areas, this gives rise to large uncertainties, which make any prediction of the current factor of safety based upon the existing measurements highly uncertain. *E.g.* Ex. CR 1 Attachment 1 at 5.

Adding to this uncertainty is the proposed approach of using a capacity reduction factor to take account of imperfections in the shape of the drywell shell. The original General Electric ("GE") study utilized an enhanced capacity reduction factor to take account of the beneficial effects of hoop stress. Citizens' Ex. 55, Report of Brookhaven National Laboratories at 3. The reviewers found that this approach may have double-counted these effects and recommended further evaluation. *Id.* at 4-5.

Similarly, Sandia reviewed the justification for using the enhanced capacity reduction factor, found that the references provided were inadequate, and so decided against using the enhanced factor. NRC Staff Ex. 6 at 67. Sandia confirmed this finding at the January 18, 2007 ACRS meeting. Transcript of ACRS Meeting on January 18, 2007 ("Tr1") *available at* ML070240433 at 284:2-11. At that meeting, the NRC Staff also concurred with Sandia. Tr1 at 288:12-19. However, at the February 1, 2007 ACRS meeting, the Staff indicated that the enhanced capacity factor was acceptable, but did not ask Sandia to provide any further presentation. On February 9, 2007, the supervisor of the Sandia study stated that Sandia's views "differ somewhat from the opinions presented by the licensee and the Staff" at the ACRS meeting on February 1, 2007. E-mail from Hessheimer to Ashley, dated February 9, 2007 available at ML070430292. The e-mail further explained that the three-dimensional model already took explicit account of the hoop stress. Thus, while it might be appropriate to enhance the capacity reduction factor when using the formulae specified in the ASME code, it is not appropriate to enhance the capacity reduction factor when using a three-dimensional finite element model. *Id.*

The choice of the capacity reduction factor makes a major difference in the predicted factors of safety. The Sandia Study used a reduction factor of 0.207, while GE used a factor of 0.326. NRC Staff Ex. 6 at 67; T1 at 96:15-17. Thus, if AmerGen predicts a factor of safety of 2.0 by multiplying the output of the three-dimensional model by 0.326, Sandia believes that the actual predicted factor of safety should be 1.27. *See* T1 at 292:25-293:9. At minimum, this shows that significant uncertainty in the choice of an appropriate capacity reduction factor adds to the already large uncertainties that arise from the limited scope and frequency of the thickness measurements and the need to make other assumptions about drywell properties and behavior. It is therefore important to ensure that the uncertainty in the capacity reduction factor is fully reflected in the sensitivity analysis.

The studies to date show that the current factor of safety is highly uncertain and is, at best, right on the edge of what is required by the ASME code. Because no amount of computer analyses can eliminate the uncertainties about the thickness and the degree to which the shape of the vessel reduces its strength, it is highly likely that any sensitivity analysis that takes full account on the current uncertainties

will indicate that there is a considerable chance of both compliance and non-compliance with the safety factor requirements, failing to establish compliance to the Required Level of Certainty.[5] If this is the case, a more refined analysis would be needed prior to any decision to extend the license.

Finally, even if the sensitivity analysis based upon the additional requirements unexpectedly showed compliance with the CLB to a high degree of certainty, Judge Baratta recognized that further deterioration of the shell could reduce the safety factor to below 2.0 while AmerGen still passes the measurements as compliant with the acceptance criteria. Additional Statement of Judge Baratta at 3. Similarly, Judge Abramson recognized during the hearing, "we don't have an analysis of how much . . . degradation this shell can take before it approaches buckling." Tr. at 510:19-21. Because there is currently no assessment of how much more corrosion, if any, would be acceptable, it is impossible to determine an appropriate monitoring scope or frequency. In addition, if future results show further thinning, there will be no way of knowing whether they are consistent with the CLB. Finally, without this limiting margin, it is impossible to determine how accurate future estimates of thickness need to be. Therefore, as AmerGen appears to have already recognized, the three dimensional analysis must also be used to determine the thickness margin above the safety factor requirements, by reducing the modeled thickness in steps until compliance is not predicted to the Required Level of Certainty. Citizens Ex. 65.

IV. State-of-the-Art Measurements And Analysis May Provide Sufficient Certainty For Licensing

Citizens anticipated the need for a more refined analysis some time ago and engaged Stress Engineering Services, Inc. ("Stress"), a well-qualified firm of structural engineers, to determine if AmerGen's commitment to evaluate future UT results in the sand bed region "per the existing program" was adequate. This material, which is attached as Ex. CR 2, was presented to the Board on July 25, 2006 in support of Citizens comprehensive contention regarding the drywell. The resumes of the individual

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[5] As Dr. Hausler points out, one approach to ensure that the sensitivity analysis fully accounts for the uncertainties in the input parameters would be to automate the analysis by assigned probability density functions to the thickness measurements and other inputs, then using a Monte Carlo simulation combined with contour plotting and extrapolation techniques to generate a probability density function for the predicted factors of safety. Ex. CR 1 Attachment 1 at 5.

engineers from Stress who provided the opinion were provided as Exhibit NC 11 (*available at* ML062140418).

In its expert opinion dated July 15, 2006, Stress pointed out that the applicable engineering code relates primarily to pressure integrity and governs construction of pressure vessels, not serviceability. Ex. CR 2 ("Stress Opinion") at 2. Thus, in stark contrast to the current situation, the code assumes that vessel thickness values are determined by design with a known small uncertainty. Because the margins in the code are designed to protect against unidentified uncertainties, Joshua M. Reinart & George E. Apostolakis, *Including model uncertainty in risk-informed decision making*, 33 Annals of Nuclear Energy 354, 355 & 357 (attached as Ex. CR 3), it is unacceptable to rely upon the code-required margin of safety to offset the uncertainty about code compliance that stems from the lack of certainty concerning the thickness of the vessel wall or the uncertainty in the shape due to operational loading or environmental degradation. *See e*-mail from Hessheimer to Ashley, dated February 9, 2007.

Stress opines that much better techniques than those used by the licensee are now available, are code compliant, and provide the most accurate assessment of vessel integrity possible. *Stress Opinion* at 3. One critical advance is the use of lasers to map the actual shapes of pressure vessels, along with sophisticated UT techniques that measure the wall thickness. *Id.* Thickness measurement techniques using other technologies have also advanced. Ex. CR 1, Attachment 1 at 5-6. Stress also points out that the G.E. analysis used idealized geometries, such as a perfect sphere for the lower part of the drywell. *Stress Opinion* at 1-2. The calculations were then adjusted by making assumptions about surface irregularities, plasticity, and local buckling using the capacity reduction factor. *Id.* at 2. Thus, the laser measurement technique described by Stress can greatly reduce the uncertainty associated with the capacity reduction factor. In this regard, AmerGen stated that it would use the "actual geometry" of the shell in its three dimensional analysis, apparently recognizing the value of this approach. Tr; 660:1-5.

Thus, a three dimensional computational analysis using input derived from state-of-the-art measurement techniques could provide a much more precise prediction of the ability of the drywell to withstand buckling. In addition, the model could then be used to predict the current margin and the

expected service life of the vessel, as anticipated by AmerGen using the current approach. Citizens' Ex. 65. Depending on the results, such a study could provide the basis for a licensing decision based upon affirmative knowledge of drywell safety now and during any extended licensing period, rather than a default decision based upon lack of knowledge. However, to facilitate detailed design of such a study, it would be ideal to have some indication from the Commission about the Required Level of Certainty.

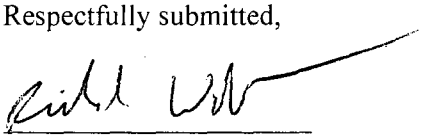
V. Reasonable Assurance Must Be Established Prior To Licensing And Should Be Established Prior To Refueling

There is no dispute that licensing may only proceed if there is reasonable assurance that the drywell shell will meet CLB requirements on April 10, 2009, the first day after the expiration of the current license, and thereafter. 10 C.F.R. § 54.29; Citizens Petition for Review at 3-4. Because such assurance is currently lacking, *id.* at 11-12, the Commission cannot decide whether to renew Oyster Creek's operating license until AmerGen submits a modeling study that establishes current and ongoing compliance with the buckling factor requirements of the CLB with reasonable assurance.[6]

CONCLUSION

For the foregoing reasons, the Commission should require AmerGen to conduct additional analyses of the factors of safety of the drywell shell prior to any decision on relicensing and grant any other relief as it may see fit.

Respectfully submitted,



Richard Webster, Esq.
Eastern Environmental Law Center
Attorneys for Citizens

Dated: June 11, 2008

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[6] Although it is concededly beyond the scope of the proceeding below, Citizens also ask the Commission to consider whether there is reasonable assurance that the drywell will comply with the requirement for a factor of safety of 2.0 during the next refueling outage, scheduled for October 2008. If further analysis is needed to establish compliance with the CLB during refueling prior to expiration of the license in less than a year, that analysis must also be needed prior to the scheduled refueling in October 2008. Citizens recognize that this issue is primarily the responsibility of the Staff, but the Commission should consider whether to exercise its supervisory authority over the Staff because the issues concerning relicensing also have implications for current safety.

EXHIBIT CR 1

EXHIBIT CR 1

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE COMMISSION

In the Matter of	)	
	)	
AMERGEN ENERGY COMPANY, LLC	)	Docket No. 50-219-LR
(Oyster Creek Nuclear Generating Station)	)	
	)	
	)	
	)	

DECLARATION OF DR. RUDOLF HAUSLER

1. My name is Dr. Rudolf Hausler. Citizens have retained me as an expert witness in proceedings concerning the application of AmerGen Energy Company LLC to renew its operating license for the Oyster Creek Nuclear Generating Station ("Oyster Creek") for twenty years beyond the current expiration date of April 9, 2009.

2. I am an expert on the corrosion of metals during operation.

3. The attached memorandum dated June 10, 2008 represents my current opinion regarding the topics it covers.

I declare under penalty of perjury that the foregoing and the attached memorandum, dated June 10, 2008 is true and correct.

Executed this 10 day of June, 2008 at Kaufman, Texas

Rudolf H. Hausler

Rudolf Hausler, PhD

Attachment 1

CORRO-CONSULTA
Rudolf H. Hausler
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Memorandum

To Richard Webster, Esq.
Legal Director
Eastern Environmental Law Center
744 Broad Street, Suite 1525
Newark, NJ, 07102

June 10, 2008

Subject: Response to NRC Commissioners' Request Regarding Additional Requirements of Judge Baratta

The Commissioners ask the parties to address the following:

> *Explain whether the structural analysis that AmerGen has committed to perform, and that is reflected in the Staff's proposed license condition, matches or bounds the sensitivity analysis that Judge Baratta would impose. In any event, explain whether additional analysis is necessary* [1].

I. Background

There is no dispute that the Drywell Liner at the Oyster Creek Nuclear Generating Station has been subject to severe corrosion in the (former) sand bed region. Several locations have been documented where the shell has lost in excess of 40 percent of its original wall thickness [2]. I have in the recent past reviewed a broad spectrum of the documentation from GPU-Nuclear/Exelon/AmerGen surrounding this problem [3], and generated a number of expert opinions and affidavits [4] based on that documentation.

I have consistently argued that the external UT data acquired in 1993 following the removal of the sand were not sufficient to fully characterize the degradation of the drywell liner while the internal UT data did also not sufficiently account for all the corrosion that had occurred in the sandbed region. This leads to considerable uncertainty in our knowledge of the current state of the sand bed region. Unfortunately, but understandably, the predictions of the residual safety factor did not attempt to estimate this uncertainty numerically.

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[1] Commissioners' Order (Requesting additional briefs), 5/28/08
[2] Original wall thickness is nominally 1.15 inches. Residual wall thickness of less than 0.7 inches has been documented.
[3] See Citizens' Exhibit B, Attachment 2 and Citizens' Exhibit C, Attachment 1
[4] See Section VI: Additional References, Anthology of Memoranda, Affidavits etc. ...

One does not therefore at this point in time know with any kind of confidence whether the drywell liner still complies with the code. Repeating the same calculations again **with the latest input data**, would not reduce the uncertainty, but could allow the uncertainty to be better quantified, by performing sensitivity analyses [5]. However, deriving some sort of confidence limits from the existing data either by refining the contour plots or similar methodology, as Judge Barrata suggests, may well lead to very large confidence intervals spanning a large range from compliance to non-compliance, which in the end would not be sufficiently determinative. The only real solution to the problem is the acquisition of additional data (in areas not previously examined by UT) that would permit a realistic representation of the corroded surface in the sandbed area, which could then be used in a 3-D model for the FEA and subsequent sensitivity study. Fortunately, advances in measurement techniques that have taken place since the existing monitoring regime was devised in 1993 have made such an approach totally realistic.

II. The Consequences of Weakening of the Shell

Although I am not a structural engineer I understand from other testimony that the factor of safety during refueling is precariously near or below the minimum of 2.0 embedded in the CLB[6]. If by modeling a safety factor of 2.15 were calculated, then the confidence limits could not be larger than +/-0.15 or 7.5%. Judge Baratta seems to be suggesting that these confidence limits, or by implication the uncertainty in additional modeling results using available measurements, could be established by a series of sensitivity analyses[7]. I believe this is a pragmatic and sensible first step, but, if the full range of uncertainty is taken into account, is unlikely to produce a definitive result.

In 2006 NRC commissioned a study by Sandia Laboratories to analyze the structural integrity of the Drywell Containment at Oyster Creek [8]. Very broadly the results showed that in the non-degraded (as built) state the safety factor in the sandbed region was 3.85 [9] while in the degraded study the estimate was 2.15 [10]. This corresponds to a reduction of structural strength of 44%. The Sandia result is based on the average remaining wall thickness in the sandbed region in each 36 degree bay, calculated using the 1992 external measurements, which on average showed thicker walls that the latest measurements. Thus, the Sandia study probably overestimated the absolute factor of safety. Furthermore, the Sandia Study did not address the uncertainty (confidence limits) in the prediction, although it did point to many sources of uncertainty and suggested that more attention should be paid to the degree of degradation rather than the absolute prediction of the factor of safety. Supporting the view that the Sandia Study probably overestimated the factor of safety, Dr. Hartzman of the NRC Staff has testified that if the contour plots

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[5] AmerGen proposes to repeat the past measurements, at the identical locations, for use in a new modeling effort. This may remove some uncertainty in the data, but does not remove the uncertainty relative to the areas that had never been measured.
[6] See ASLB Initial Decision 12/18/07, page 19
[7] See ASLB Initial Decision 12/18/07: Additional Statement by Admin. Judge Anthony J Baratta, PhD.
[8] Sandia Report Sand207-055: Structural Integrity Analysis of the Degraded Drywell Containment at the Oyster Creek Nuclear Generating Station, Jan. 2007
[9] Sandia Study (NRC Staff Ex. 6) at 70
[10] Sandia Study page (NRC Staff Ex. 6) at 79

of the measured areas that I produced are correct, the factor of safety would be approximately 1.9 [11].

III. The Nature of the Available Corrosion Data [12]

In order to perform not only the structural integrity calculations, but the sensitivity analysis as well, one needs to discuss first the uncertainties underlying the available data.

Before discussing our actual knowledge of the degree of corrosion damage of the drywell liner a fundamental principal of confidence interval (or confidence limits) needs to be understood. It is entirely reasonable (and customary) to assess the quality of a bin of say a thousand bolts by sampling a few, measuring the property of interest of the few, and defining from the results the average property of the lot as well as the probability that the individual will vary from the average by an amount given by the confidence limits defined by the desirable degree of confidence – 95%, 99% or even better. This is so because the bolts all were manufactured by the same machine and can therefore be assumed to belong to the same family of bolts, which can be characterized by a mean and a probability density customarily referred to as the Gaussian distribution curve.

It is difficult to apply this principle to the distribution of corrosion in the sandbed area for two main reasons. First, not all of the longitudinal sections (called bays) belonging to the former sandbed area have suffered the same degree of corrosion. Second it has been assumed that in the bays, where severe corrosion has been observed, these areas have been sufficiently characterized without the need to characterize the entire bay. In other words it had been assumed that the areas examined represented all the corrosion there was.

Because of this state of the current knowledge and related assumptions one needs to deal in depth with the following three concerns:

- The measurements themselves (how accurate and reproducible are the measurements?)
- The paucity of measurements (are the few available measurements really representative of the state of the degraded shell?)
- How does one need to deal with the external point measurements (see below)

<u>Accuracy and Reproducibility:</u>

What is known about the residual wall thickness of the dry well liner has been obtained by means of ultrasonic thickness (UT) measurements. The accuracy (uncertainty) of UT measurements has basically two origins. The instrument and the operator account for a standard deviation of about 1.5to 2% of wall thickness for a single measurement (modern instruments may be slightly more accurate). If one adds in the reproducibility of

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[11] NRC Staff response to initial Presentation and Response to Board questions, Aug 17, 2007, A28

[12] The issues discussed here have been detailed in a number of previous submissions either as affidavits or direct testimony; (See list of submissions attached as Appendix A)

performing the same measurement repeatedly over time the overall standard deviation becomes 0.028 inches or in this case 3.5% of wall thickness. This means that for single measurements [13] the 95% confidence limits are of the order of +/- 7% of wall thickness. For example, for a residual wall thickness of 800 mils, this would put the 95% confidence interval from 856 to 744 mils [14].

Two types of measurements have been made over the years: a) internal UT measurements [15] to monitor progressive corrosion and b) external UT measurements after the sandbed had been removed to assess the remaining integrity at points that were often located at visually identified minima in the external surface.

The Density of Measurements:

The internal measurements were made by using a 6 inch by 6 inch template with holes in it every inch so that within the 36 square inches 49 measurements could be made repeatedly over the years at the same spots. Considering that the surface area of the sandbed region for each bay is 73 square feet [16] it is unlikely that that measuring 0.3% (36 square inches is 1/4 ft^2) of this area on top of the sandbed would be representative of the entire sandbed, unless the corrosion in the sandbed were distributed uniformly. Analysis of the data and the visual inspections show that the occurrence of corrosion is actually highly variable, as AmerGen's [17] testimony regarding external corrosion measurements confirms (see below).

A large number of measurements in a small are only improves the information about the small area but says nothing about the areas beyond.

The only quantitative knowledge of the areas beyond the grids comes from the external measurements. These measurements were made after the sand had been removed from the sandbed in order to confirm the extent of corrosion. It was established that in several instances corrosion damage increased in severity deeper down in the sandbed, thus showing that the internal grid data cannot be representative of the entire region. Again, the extent of the external UT measurements covered an area of varyingly about 10 - 30 % of any one bay.

Typical results for Bay 1are shown in Figure 1 of Citizens Exhibit C1, Attachment 1[18]. The area covered by the measurements is 27 ft^2 or about 1/3 of the total area of the bay

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[13] AmerGen has very rarely repeated measurements in order to improve their accuracy.
[14] It should be noted that neither AmerGen nor its predecessor ever examined the accuracy and/or reproducibility of the UT measurements.
[15] Ultrasonic wall thickness measurements
[16] NRC Staff Ex. C at A28
[17] AmerGen Ex. B Part 3 at A18: "portions of the exterior surface of the drywell shell of the sandbed region . . . have a very uneven surface." Clearly corrosion in the sandbed region was not uniform.
[18] The Figure shows the area where measurements had been made. Each measurement is represented by the black square. The vertical and horizontal coordinates are in inches from a reference point roughly located at the top of the sandbed. The values of the measurements (residual wall thickness) have been translated to contour lines representing the average residual wall thickness within the area contoured by the plots.

and it is clear that corrosion could have extended beyond –60 inches and further into the non-measured areas. Furthermore, corrosion did not occur only near the top of the sandbed (the reference point for the top of the sandbed being roughly at 0 inches). Because the density of measurements is very low and clustered around two areas it is clearly difficult to depict what the surface might have really looked like. Drawing contour plots as shown was one way of presenting the data. However, the algorithm used to generate contour plots essentially calculates average wall thicknesses between one point and all the surrounding ones. No consideration is given to the fact that each measurement only represents the residual wall thickness at a specific point (z-direction), and no information has been gained about how far the degradation extends in the x-y dimensions. These are equally important with regards to integrity. This is probably why Judge Barrata suggested to perform a series of sensitivity analyses using a methodology similar (but not necessarily identical) to the one used to generate the contour plots.

IV. The Sensitivity Analysis

There are a number of crucial inputs to the structural integrity analysis of the degraded Drywell containment. These deal with:

- Material properties;
- The geometry of the vessel;
- The extent of the degradation described above.

The sensitivity analysis aims at determining the degree of variation of the safety factor as a function of uncertainties in the any of the above parameters. With respect the extent of degradation for which standard deviations and confidence limits have been established it should be possible to define probability densities for each measured point. This should be done for the external measurements in particular because they are single point measurements that are quite variable. By means of a Monte Carlo simulation combined with the extrapolation techniques I have proposed (or any other applicable statistical stochastic procedure), a series of sensitivity analyses could be calculated, which could be used to estimate the lower 95% confidence interval for the output.

It should be noted, however, that in light of the few data points, absence of information of corrosion in the x-y directions, the missing information about a large extent of the surface area, and the lack of any measurements of the actual geometry, the confidence limits will very likely be large and perhaps unreasonably so. That may mean that the lower confidence interval of the predicted factor of safety could be very much below two, leading to an indeterminate result.

V. How to approach the Problem

For all of the above reasons we think that the question of whether the drywell meets the safety requirements can only be resolved by taking additional measurements in the areas of interest not previously explored in detail. Fortunately, since 1993, when the current

measurement regime was devised, technology has advanced considerably. It should be possible to actually map the residual wall thickness over the entire area of the individual bays by means of automated instruments using UT, MFL (magnetic flux leakage), PEC (Pulsed Eddy Current) or a combination of these techniques [19]. The digitized maps of the surface thickness could then be used directly as input to the finite element analysis of the residual strength of the vessel.

In addition, I understand that Stress Engineering Services has proposed using a laser technique that could measure the shape of the vessel.

VI. Additional References (Anthology of Memoranda, Affidavits, etc. in Support of Citizens' Contention re. Oyster Creek Dry Well Corrosion)

- Nov. 10, 2005, Memorandum R. H. Hausler to Paul Gunter, NIRS, re. *Oyster Creek Drywell Liner Corrosion*; in support of Citizens' Request for Hearing and Petition to Intervene, Nov 14, 2005
- Feb. 4, 2006, Memorandum R.H. Hausler to Paul Gunter, Richard Webster, Esq., re. *Oyster Creek Drywell Liner Corrosion,*
- March 16, 2006, Memorandum R. H. Hausler to Richard Webster, Esq., Paul Gunter, re. *Oyster Creek Dry Well Corrosion: Additional Evidence for Continued Corrosion*
- May 3, 2006, Memorandum R.H. Hausler to Richard Webster, Esq., Paul Gunter, Oyster Creek Dry Well Corrosion: Comments re. *"Audit Q&A (Questions No. AMP –141, 210, 356) dated 4/5/06 Ref. ML060960563*
- June 9, 2006, Memorandum R. H. Hausler to Richard Webster, Esq., re. *Statistical Analysis of AmerGen-Exelon UT Data, Dry Well Corrosion,*
- June 12, 2006 Memorandum R.H. Hausler to Richard Webster, Esq., re *Discussion of Corrosion Monitoring Methodologies at Oyster Creek Nuclear Plant Dry Well*
- June 20, 2006 Memorandum R.H. Hausler to Richard Webster, Esq., re. *Oyster Creek: Pitting Evaluation of Dry Well Liner in Sandbed Region*
- June 22, 2006, Memorandum R.H. Haulser to Richard Webster, Esq., re. *Discussion of Corrosion Monitoring Methodologies at Oyster Creek Nuclear Plant Drywell*
- April 25, 2007, Citizen Exhibit B, Attach. 3, Memorandum to Richard Webster, Esq., *Update of Current Knowledge regarding the State Integrity of OCNGS Drywell Liner and Comments Pertaining to the Aging Management Thereof.*
- July 18, 2007, Citizen Exhibit B, Attach. 4, Memorandum to Richard Webster, Esq., *Review of Fitness for Service Assessment of Oyster Creek Drywell on Basis of Extended Data Analysis*
- Aug 16, 2007, Citizen Exhibit C, Attach. 2, Memorandum, *Response to ASLB re Questions concerning Statistics*

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19) e.g. Monitoring Average Wall Thickness of Insulated or difficult to access objects with pulsed Eddy Currrrent, by R. Scottini, H.J. Quakkelsteijn – RDT Group, The Netherlands IV Pan-American Conference on NDT, Buenos Aires, October 2007 (attached)

- Aug 16, 2007, Citizen Exhibit C, Attach. 3, Memorandum R. H. Hausler, to Richard Webster Esq., *Further Discussion of the nature of the Corroded Surface and the Residual Wall Thickness of the Oyster Creek Drywell.*
- Sep 13, 2007, Citizen Exhibit C-1, Attach. 1, Memorandum R. H. Hausler to Richard Webster, Esq., *Further Discussion of the External Corrosion on the Drywell Shell in the Sandbed Region.*

IV Conferencia Panamericana de END
Buenos Aires – Octubre 2007

AAENDE
Asociación Argentina de Ensayos No Destructivos y Estructuales

Monitoring Average Wall Thickness of insulated or difficult to access objects with Pulsed Eddy Currrent

R. Scottini, H.J. Quakkelsteijn – RTD Group, The Netherlands

Abstract

The method of Pulsed Eddy Current (PEC) has been successfully applied in corrosion detection for several years now. Whereas field experience on insulated objects has grown significantly, the technique's characteristics make it also highly suitable for other field situations where the object surface is rough or inaccessible. Because (surface) preparations can be avoided the tool provides a fast and cost-effective solution for corrosion detection. Due to the high repeatability accuracy PEC technology is specially of interest for monitoring purposes.

An overview of the fundamentals and the INCOTEST® pulsed eddy current tool for corrosion detection is presented and application ranges are discussed. Several field applications other than insulated objects are presented. These range from the inspection of objects covered fire proofing, to rough or corroded surfaces, coated objects and objects covered with marine growth.

These spin-offs offer interesting possibilities in many areas of industry such as sub sea piping, offshore jackets, civil engineering and FPSO ship hull inspection.

Introduction

Corrosion under insulation is a major concern for the owners and operators of almost all carbon steel installations and structures. Periodic or continuous inspection of objects for occurrence of corrosion or monitoring the extent and severity of known corrosion areas should ensure operation of the installation within the safe zone.

To operate the installation at minimum cost, new techniques can be applied to minimise the overall maintenance and inspection costs. Such techniques can aim at reducing the total number of activities either by reducing the number of selected areas to look after or by reducing the overall costs per inspected area. The latter, for instance, is possible by reducing the peripheral costs of inspection (preparation, cleaning, access etc.).

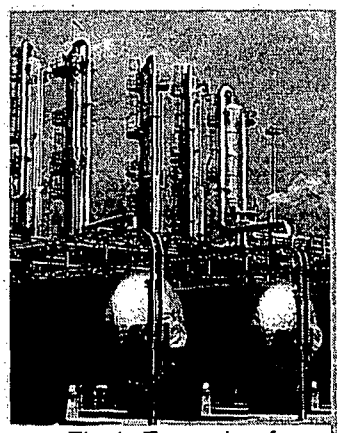
Fig 1: Example of insulated objects.

The INCOTEST pulsed eddy current tool can assist by bringing down both the number of selected areas and the peripheral cost in several applications.

This tool was developed for the detection of corrosion under insulation (CUI). It allows the detection of wall thinning areas without removing the insulation. Using this tool to indicate the affected areas can lead to significant cost reduction. Fewer areas need follow-up and less insulation needs to be removed. Also, in case of asbestos insulation the safety hazards are diminished.

INCOTEST applies pulsed eddy currents for the detection of corrosion areas. A pulsed eddy current technique uses a stepped or pulsed input signal, whereas conventional eddy currents use a continuous signal. The advantages of the pulsed eddy current technique are its larger penetration depth, relative insensibility to lift-off and the possibility to obtain a quantitative measurement result for wall thickness.

This leads to the characteristic which makes it suitable for the detection of CUI: no direct surface contact between the probe and the object is necessary. Also, this tool can be employed in other field situations where the object surface is rough or inaccessible. After a brief introduction of the theory, some of these applications are discussed.

Pulsed Eddy Currents for corrosion detection

The applied operating principle of pulsed eddy currents can vary from system to system. In order to obtain a quantitative reading for wall thickness INCOTEST uses a patented algorithm that relates the diffusive behaviour in time to the material properties and the wall thickness. It operates only on low alloy carbon steel.

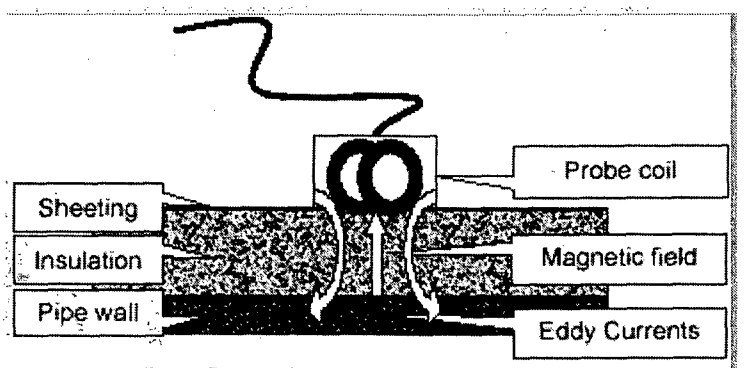

Fig 2: Principle of operation INCOTEST.

The principle of operation is illustrated in Figure 2. A pulsed magnetic field is sent by the probe coil. This penetrates through any non-magnetic material between the probe and the object under inspection (e.g. insulation material). The varying magnetic field will induce eddy currents on the surface of the object. The diffusive behaviour of these eddy currents is related to the material properties and the wall thickness of the object.

The detected eddy current signal is processed and compared to a reference signal. The material properties are eliminated and a reading for the average wall thickness within magnetic field area results. One reading takes a couple of seconds. The signal is logged and can be retrieved for later comparison in a monitoring approach.

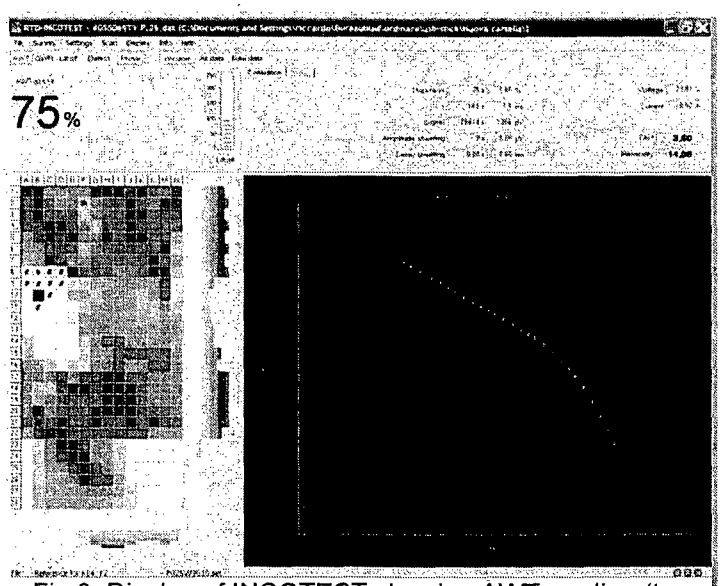

Fig 3: Display of INCOTEST showing AWT reading (top left), logged inspection grid (bottom left) and the decay of the eddy currents (bottom right).

The area over which a measurement is taken is referred to as the footprint. Probe design is such that the magnetic field focuses on an area on the surface of the object. The result of the measurement is a reading of the average wall thickness over this footprint area. The size of this area is dependent on the insulation and object thickness, as well as the probe design. Roughly, the footprint can be considered to be in the order of the insulation thickness. Due to the averaging effect, detection of highly localised defects types like pitting is not reliable with this tool. This effect is illustrated in Figure 4.

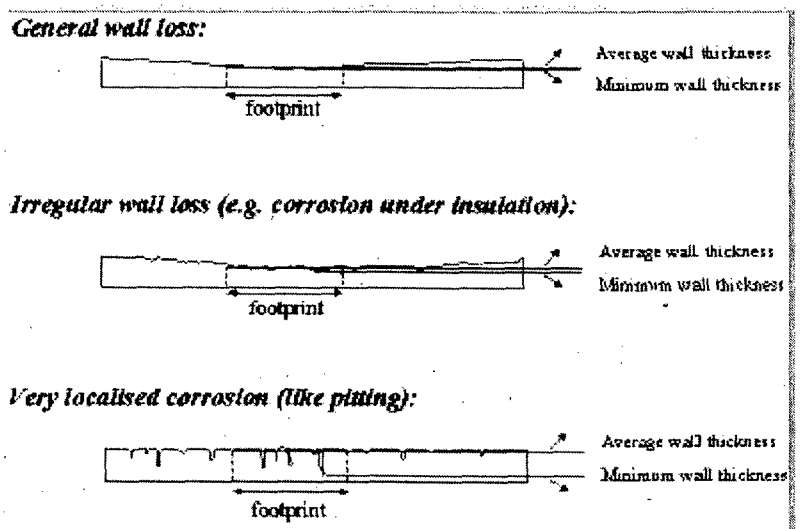

Fig 4: Difference between average and minimum WT within the footprint area.

Although the average wall thickness reading is not a direct replacement of the commonly used UT obtained minimum wall thickness a quantitative result is obtained that can be interpreted unambiguously.

The outer application ranges of the INCOTEST tool can be described by:

- Low alloy carbon steel
- Pipe diameter > 50 mm or 2"
- Nominal wall thickness between 6 mm and 65 mm
- Insulation thickness up to 200 mm
- Sheeting thickness up to 1 mm stainless steel, aluminum, galvanised steel
- Object temperature > -100°C to < +500°C

These ranges are determined on condition that a reliable signal can be obtained under regular field conditions.

Inspection approach

As with any other NDT technique, the pulsed eddy current technique has its own merits and cannot be a direct substitute for an existing NDT technique in an existing NDT inspection program. The characteristics of INCOTEST result in the application of the tool with various intentions. Firstly, the reduction of surface preparations may be an incentive to use the tool. No cleaning, grinding or removal of coating and insulation is required.

Secondly, on-stream screening for corrosion areas can be the objective. This means detecting defects is more important than sizing them accurately. It may be done to bring some ranking in a large number of structures or objects that would otherwise not get any attention because conventional inspection is too costly. Another application can be to select areas for follow-up. For instance, in a pre-shutdown inspection the items that need follow-up during a shutdown can be identified.

On-stream monitoring of corrosion areas using INCOTEST is another approach that is of interest because of intrusion on the process is kept to a minimum. The data of previous measurements can easily be retrieved and compared.

Finally, in a risk based inspection approach a choice is made for the level of information required and the necessary certainty for inspection of a particular object. This leads to a choice for a non-destructive testing approach in which pulsed eddy current can be one technique.

Field applications

Fire proofing

Many foundations in installations, such as skirts of process columns and the supports of spherical storage tanks, are covered with a layer of fireproofing for obvious safety reasons. Small cracks or damages to the fireproofing may cause ingress of water, resulting in corrosion underneath the covering. The deterioration process can not readily be detected from the outside. Failing adequate condition monitoring, the deterioration process may eventually cause the object foundation to collapse with disastrous results.

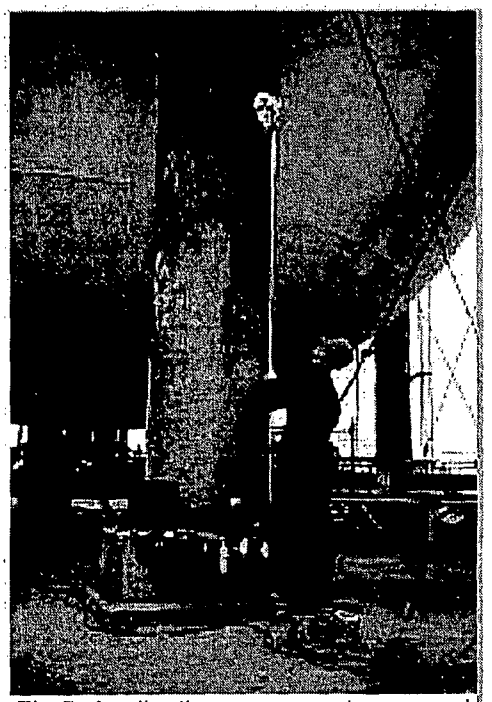

Fig 5: Application on concrete covered object: support legs of spherical storage tanks.

As these fire proofing materials are non-magnetic and non-conducting, the magnetic field can freely propagate between the probe and the object under inspection. Hence, pulsed eddy current can be used to detect corrosion areas without removing the fire proofing material.

To obtain a picture of the foundation's condition, measurements are taken in several points of a defined grid. On the supports of spherical tanks a rapid screening is done by taking readings on four wind directions distanced 100mm-150mm apart axially and starting 300mm from the foot. This results in about 100 readings per support leg. In one inspection day all eight support legs of a tank can be screened and reported.

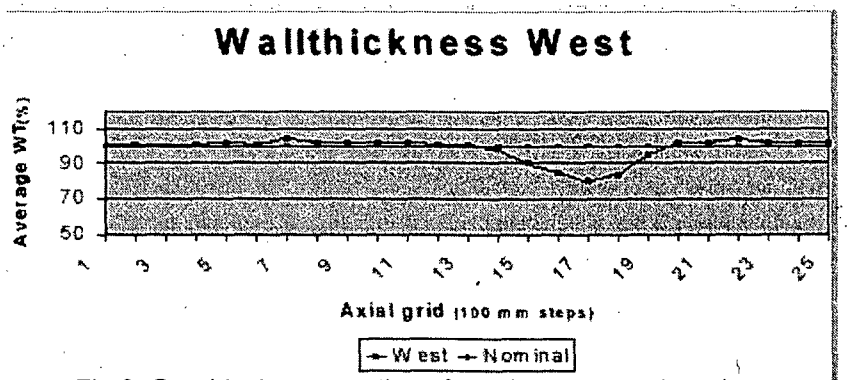

Fig 6: Graphical presentation of results on one sphere leg.

All average values measured are presented in a table together with a graph of the results indicating areas of interest for further action. These results can be used for strength calculations indicating the necessity whether or not to take action on the support leg.

Again, using INCOTEST the owners/operators of these structures can find out the current condition in a rapid and cost-effective manner.

Sub sea applications

Many bank-protections, ports and waterworks in areas with a soft soil consist of steel sheet pilings. These sheet pilings have only a very limited protection against the elements. As a result the unshielded steel surface will be attacked by various forms of corrosion, among Accelerated Low Water Corrosion (ALWC). Similar situations occur for instance at risers and the support pillars of jetties. In all these situations both time and money are saved by using the ability of pulsed eddy current to penetrate dirt and marine growth

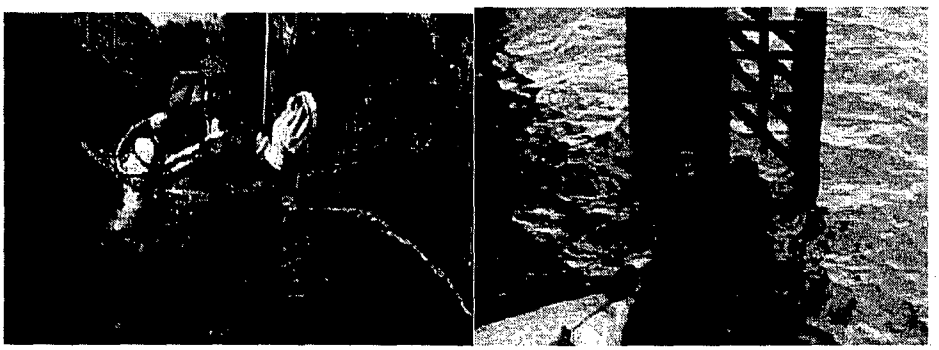

Fig 7: underwater and through marine growth: sheet piling and jetty

Maintenance including coating the surface is a costly action. The need to create a clean and dry environment below water level, and in the tide zone is, the most expensive .

The conventionally used methods of UT require extensive cleaning. Because no cleaning is necessary the use of INCOTEST in this situation leads to a faster inspection. The inspection can be carried out both above and below water level. Based on this result further maintenance can be done creating only localised clean and dry areas.

Even more inaccessible areas are sub sea piping and ship hulls of FPSO's. Many sub sea piping is covered with coatings or concrete. Inspection of these pipe lines is mainly done with intelligent pigs, however that requires that the line is taken out of service. Inspection of INCOTEST from the outside is possible with the use of divers or a ROV. Main aim it to obtain inspection data, without the associated inspection costs, in order to decide if immediate follow up is necessary or to determine the next inspection interval.

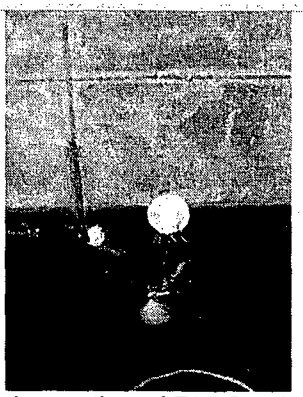

Fig 8: Inspection of FPSO ship hull.

Monitoring

Another advantage is also apparent in the relatively small lift-off ranges of several millimeters coating material or directly on the object itself. The repeatability of INCOTEST is 2% and thus an excellent tool for monitoring. Although UT is very accurate, it is commonly known that the repeatability of UT is relative poor.

Once a defect is identified, e.g. by mapping with UT to determine the weakest point, a INCOTEST probe is positioned and measures every day, week or month. At PEMEX, The "Ing. Antonio M. Amor" refinery in Salamanca, Mexico encountered an interesting application for INCOTEST;

Pemex performed a lot of experimental inspections in 2005. Inspection programs on insulated tank walls were established where, in the past, no inspections could be performed because of accessibility problems. During a shutdown a severe wall loss was detected in the shell of a heat exchanger containing Hydrochloric acid. Because a spare was not available, and in order to maintain safe production, Pemex found an interesting solution. "After basic repairs, Pemex decided to monitor the shell thickness by performing INCOTEST with an interval of 15 days. Due to the coating and operating temperature it was not possible to perform UT. A wall reduction of 16% was detected in only a few weeks. However, Pemex was able to continue production under safe conditions until a replacement was available. Pemex gained a lot of experience, which will be implemented in our inspection plans", says Ing. Jorge Galvan Pena, Superintendente de Inspeccion Tecnica

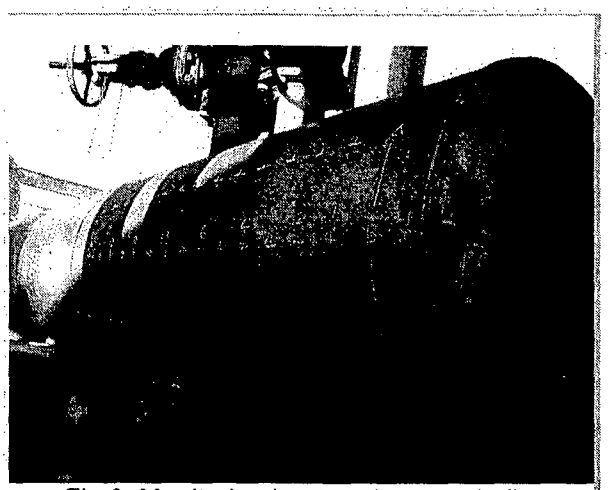

Fig 9: Monitoring heat exchanger shells.

Conclusion

Beside insulated objects INCOTEST proves a suitable application for situations where access to or preparation of the object surface is hampered. The application of this pulsed eddy current technique can be done with several different inspection approaches. Because of its unique characteristics it can play an important role in the inspection strategy or RBI approach of an entire installation. Practical examples have been given for situations where dirt, corrosion, water, concrete or coating material hamper direct surface access. Because (surface) preparations can be avoided the tool can provide a fast and cost-effective solution for corrosion detection.

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About the authors

Mr. Scottini earned his masters in material engineering at the University of Trento, Italy. Since 1997 Mr. Scottini is employed by RTD Group and has been actively involved in the technical development of INCOTEST and the development of new applications. Mr. Scottini is considered the leading technical expert on PEC.

Mr. Quakkelsteijn earned his bachelor in Technical Business Administration at the TH-Rijswijk, The Netherlands. Since 1997 Mr. Quakkelsteijn is employed in RTD Group and has been involved on on-stream leak tightness testing, advanced ultrasonic tube testing and is now overall responsible for INCOTEST in the RTD Group.

EXHIBIT CR 2

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July 15, 2006

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SES Project No.: 131377

Subject: Cursory Check of Structural Analyses, Oyster Creek Drywell Vessel

Dear Mr. Webster:

Recently, you requested that Stress Engineering Services, Inc. consider several documents that you provided and others that were made available to us through internet link references from the U. S. Nuclear Regulatory Commission. These documents concern the license renewal of the Oyster Creek Nuclear Generating Station.

One issue of contention in the license renewal at hand is whether the corroded drywell shell retains adequate strength for continued service. Your specific instructions were to review the structural analyses and comment on the approach used to assess their adequacy. Thus, we did not address any issues related to either the preexisting corrosion damage or potential ongoing corrosion of the vessel, unless it was salient to our review of the structural analysis work.

This report contains two sections. The first section addresses the general structural analysis methods and results. The second section addresses the ASME Code provisions. In both sections, it is important to note that our comments and opinions are based on a severely limited review that only touches the highlights of the respective subjects. A more detailed review is needed to address these subjects with the depth of study necessary to uncover the fundamental differences between the work that was done in support of the license and the state-of-the-art in structural analysis.

Structural Analyses

At issue is the structural adequacy of the drywell shell, which has the shape of an inverted light bulb. The primary structural concern is the drywell shell's ability to resist buckling with an adequate margin for continued safe operation.

The structural analysis results offered by AmerGen were obtained using typical techniques for the period of time in which the analyses were performed. Due to the limited computational power that was readily available at the time, the computer-aided analysis performed by General Electric (GE) utilized relatively small slices of

the vessel, idealized geometries (perfect spheres, cylinders, etc.), and required computationally efficient calculation techniques. Calculated buckling load behaviors for the idealized geometries were subsequently adjusted using assumptions or "capacity reduction factors" for surface irregularities, plasticity, and local buckling; and the resulting adjusted values were taken as representative of the actual buckling load. GE compared the calculated buckling loads with the imposed loads, and safety margins were determined for comparison to ASME Code minimum requirements. Primarily because of these computational limitations, the finite element analysis performed by GE on the drywall vessel may not be adequate to capture its global behavior, which may be some combination of symmetrical and anti-symmetrical buckling.

The state-of-the-art has progressed far beyond the methods available to structural analysts in the early 1990s. Today, when reconstructing or reverse engineering existing structures, it is routine to use laser devices to generate "point clouds" that fully define the surfaces of pressure vessels, including any irregularities. The point clouds are digitalized, and the digitized information is converted into a mathematical representation of the actual surface shape, which is subsequently utilized for full three-dimensional modeling. Since the resulting models account for actual surface waviness, unevenness, bulges, facets, and other potentially deleterious geometric surface conditions, there is no longer any need to resort to the use of "capacity reduction factors" to determine buckling loads, as the GE analysts were forced to do.

The digitized surface is converted into a form suitable for meshing and further processing using finite element analysis (FEA). The mesh areas are then assigned the corroded thicknesses at the specific areas where they actually occur, and any future corrosion allowance is subtracted from the thickness at this time. The FEA mesh density would then be generated as fine as needed to capture the stiffness that resists buckling. The simulated loads are then applied and the buckling load and shape are directly calculated without needing imposed perturbations or anything except the measured geometry and thicknesses.

Utilization of point cloud surface mapping techniques along with measurements that represent the actual wall thickness is thought to give the most accurate structural analysis results possible, with the fewest assumptions, using current technology. Three-dimensional thin shell analyses can be done today with few assumptions concerning stiffness and in a way that complies with Case N-284-1-1320.

ASME Code[1] Provisions

At issue is whether the Code is the best tool available for determining the drywell's fitness for continued service.

In general, the Code establishes rules of safety relating only to pressure integrity and governing the construction[2] of boilers, pressure vessels, transport tanks, and nuclear components. Its

[1] ASME Boiler and Pressure Vessel Code, Section III, *Nuclear Components*, and Section VIII, *Rules for Construction of Pressure Vessels*, American Society of Mechanical Engineers, Three Park Avenue, New York, NY 10016

[2] *Construction*, as used in the Code, is an all-inclusive term comprising materials, design, fabrication, examination, inspection, testing, certification, and pressure relief.

wording allows for some latitude in design and analysis methods, anticipates that deterioration of pressure vessels will occur, requires the use of engineering judgment, and recognizes the inevitability of technological progress in design and analysis methods. The following statements, which we excerpted from the FOREWORD of the current edition of the ASME Boiler and Pressure Vessel Code, support this contention.

"The Committee's function is to establish rules of safety, relating only to pressure integrity, governing the construction of boilers, pressure vessels, transport tanks and nuclear components, and inservice inspection for pressure integrity of nuclear components and transport tanks, and to interpret these rules when questions arise regarding their intent... With few exceptions, these rules do not, of practical necessity, reflect the likelihood and consequences of deterioration in service relating to specific service fluids or external operating environments. Recognizing this, the Committee has approved a wide variety of construction rules in this Section to allow the user or his designee to select those which will provide a pressure vessel having a margin for deterioration in service so as to give a reasonably long, safe period of usefulness. Accordingly, it is not intended that this Section be used as a design handbook; rather, engineering judgment must be employed in the selection of those sets of Code rules suitable to any specific service or need... The Committee recognizes that tools and techniques used for design and analysis change as technology progresses and expects engineers to use good judgment in the application of these tools."

Clearly, the authors of the Code never intended that its rules be used as the only arbiter of pressure vessel structural integrity. Neither did the authors intend the rules be used to extend, possibly unreasonably, the useful life a significantly corroded nuclear pressure vessel such as the drywell. Nonetheless, some continue to rely on Code construction rules for these purposes. They continue to do so despite the existence of tools such as three-dimensional thin shell analysis that have proven to be more than adequate for nuclear applications when applied in the presence of seasoned engineering judgment.

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Available online at www.sciencedirect.com
SCIENCE DIRECT®
Annals of Nuclear Energy 33 (2006) 354–369
annals of NUCLEAR ENERGY
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Including model uncertainty in risk-informed decision making

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Received 3 August 2005; received in revised form 28 November 2005; accepted 28 November 2005
Available online 19 January 2006

Abstract

Model uncertainties can have a significant impact on decisions regarding licensing basis changes. We present a methodology to identify basic events in the risk assessment that have the potential to change the decision and are known to have significant model uncertainties. Because we work with basic event probabilities, this methodology is not appropriate for analyzing uncertainties that cause a structural change to the model, such as success criteria. We use the risk achievement worth (RAW) importance measure with respect to both the core damage frequency (CDF) and the change in core damage frequency (ΔCDF) to identify potentially important basic events. We cross-check these with generically important model uncertainties. Then, sensitivity analysis is performed on the basic event probabilities, which are used as a proxy for the model parameters, to determine how much error in these probabilities would need to be present in order to impact the decision. A previously submitted licensing basis change is used as a case study. Analysis using the SAPHIRE program identifies 20 basic events as important, four of which have model uncertainties that have been identified in the literature as generally important. The decision is fairly insensitive to uncertainties in these basic events. In three of these cases, one would need to show that model uncertainties would lead to basic event probabilities that would be between two and four orders of magnitude larger than modeled in the risk assessment before they would become important to the decision. More detailed analysis would be required to determine whether these higher probabilities are reasonable. Methods to perform this analysis from the literature are reviewed and an example is demonstrated using the case study.
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1. Introduction

Very low-probability, high-consequence events are the focus of reactor safety studies. Because of the limited number of these events, there are large uncertainties regarding their probabilities of occurrence. Uncertainty in the core damage frequency (CDF) and large early release frequency (LERF) can be separated into three classifications; parameter uncertainty, model uncertainty, and completeness uncertainty. This paper describes a methodology for the identification of basic events which have the potential to impact licensing basis decisions. We concentrate on applications of Level I, at-power, internal-events probabilistic risk assessments (PRAs) and on the decision-making process related to licensing basis changes. Once these basic events are identified, sensitivity studies are performed to determine by how much the probability of each event must be increased to have an impact on the decision. Analysis must then be done to determine whether this increase is reasonable.

PRAs use models of a plant's structures, systems, and components (SSCs) to determine the probability of occurrence of various events. Sometimes, however, there is no consensus on the appropriate model to be used. It may be that, because the system or process is not sufficiently understood, there are differing opinions as to which model most accurately represents the system. This creates uncertainty in the model, which could be related to the structure of the model, or its numerical assessments. This uncertainty in the accuracy of the model introduces uncertainty in the output of the model and, therefore, uncertainty in the output of the PRA. It is this uncertainty that we refer to in this paper as model uncertainty (Mosleh et al., 1993).

Nuclear power plant licensees may use PRA information to apply for plant-specific licensing basis (LB) changes.

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0306-4549/$ - see front matter © 2006 Elsevier Ltd. All rights reserved.
doi:10.1016/j.anucene.2005.11.010

Guidance for doing this is provided in regulatory guide (RG) 1.174 (USNRC, 2002) and includes a requirement to meet acceptance guidelines based on the plant's CDF and LERF and the corresponding changes, ΔCDF and ΔLERF, resulting from the requested change. RG 1.174 and the regulatory guidance on the definition and treatment of model uncertainty will be discussed in Section 2 of this paper. Model uncertainty is then discussed in detail in Section 3. The intent is to establish a clear understanding of model uncertainty, its interpretation, theory, and how it is handled in practice, as well as a review and discussion of model uncertainties identified as generally important in the literature. The problem then becomes how to determine which uncertainties can affect the decision. The proposed methodology for this is presented in Section 4.

This methodology begins with using the PRA of the plant to determine the risk achievement worth (RAW) importance measure (Cheok et al., 1998) of each basic event. The use of RAW to determine the effect of an event on CDF is well understood. We use RAW in the same way, but we also evaluate RAW with respect to ΔCDF. From the importance measure information, we determine which basic events could possibly affect the decision either through CDF, ΔCDF or both.

Basic events that are identified as potentially important through the RAW analysis are cross-checked with those that have been identified in the literature as having generally important model uncertainty. This cross-check results in a list of basic events that have uncertainties that could possibly affect the decision. These basic events are then analyzed qualitatively and quantitatively to determine their impact on the specific decision.

The benefit of this methodology is that important basic events are identified with respect to the change-specific decision at hand rather than through the use of a general definition of importance. This methodology also reduces the number of uncertainties that must be analyzed extensively by allowing qualitative arguments based on change-specific conditions. In its present form, the proposed methodology deals with the uncertainties associated with the modeling of events that appear in the PRA; it does not deal with model uncertainty that may affect the logical structure of the PRA itself.

In order to illustrate the proposed method, a case study is provided in Section 5. In Section 6, we evaluate the important basic events from the case study. This helps us to understand how important the uncertainty might be in the context of this case study and whether it warrants an in depth review of its probability and the assumptions underlying the calculation of its probability. Finally, Section 7 contains a summary and conclusions from this research.

2. Risk-informed decision making

The US Nuclear Regulatory Commission (NRC) encourages the use of PRA methods where practical, consistent with the state-of-the-art, to support a risk-informed regulatory framework (USNRC, 1995). RG 1.174 is a key document in this framework. It presents five principles of risk-informed decision making to be used for making decisions regarding plant-specific changes to the licensing basis. These principles are:

1. The proposed change meets the current regulations unless it is explicitly related to a requested exemption.
2. The proposed change is consistent with the defense-in-depth philosophy.
3. The proposed change maintains sufficient safety margins.
4. If the proposed change increases risk, the increase should be small.
5. The impact of the proposed change should be monitored using performance measurement strategies.

The first principle makes it clear that existing regulations not related to the requested change must still be met. The second and third principles account for some of the completeness uncertainty that exists when assessing nuclear power plant risk. This uncertainty is referred to as the "unknown unknowns," or the uncertainties that exist but have not been identified. Because the uncertainty has not been identified, it cannot be quantified. Traditional defense-in-depth measures and safety margins (the "structuralist" approach to safety (Sorensen et al., 1999)) are requirements designed to protect against these uncertainties. The fourth principle is the one we are concerned with in this paper, requiring risk increases to be small. Risk and risk increases are quantified using PRA. The fifth principle ensures that the results of a licensing base change are as expected. Performance monitoring and measurement after the change provide feedback that can be used to identify and correct unexpected problems that result from the change and also to inform future changes. Also, the effects of uncertainty on all of these principles must be considered, whether the uncertainty is explicitly included in the model or not.

Risk increases must be small. Small changes are defined using the CDF and LERF acceptance guidelines of Figs. 1 and 2, respectively. The values of CDF, ΔCDF, LERF, and

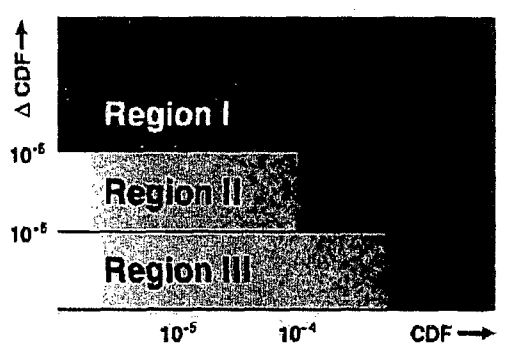

Fig. 1. CDF acceptance guidelines.

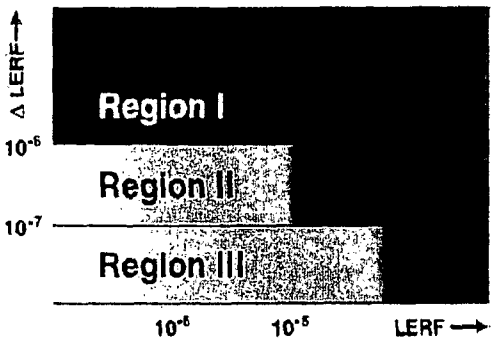

Fig. 2. LERF acceptance guidelines.

ΔLERF to be used in these figures are supposed to be the mean values of the uncertainty distributions of these quantities. This paper focuses on Level I PRAs, so we will focus on the CDF/ΔCDF guidelines. Referring to Fig. 1, the horizontal axis represents the baseline CDF. This is the frequency at which core damage is expected to occur at the plant if no plant changes are made. The vertical axis represents the ΔCDF, the amount that the CDF is expected to increase, if the proposed LB change were made.

Each nuclear power plant has an associated plant-specific CDF. Each LB change that a plant desires to make has an associated plant-specific and change-specific ΔCDF. These two risk metrics place a proposed change in one of the three labeled regions in Fig. 1. Uncertainty in the risk metric calculations prevents an exact placement of a change onto one of the three regions. Therefore, the values representing the dividing lines between the regions must be viewed as indicative, rather than definitive. Changes that have a ΔCDF placing them in Region III are classified as having a very small increase in risk and may be approved without the need for a quantification of CDF. Changes that are in Region II are classified as having a small increase in risk and may be approved, but may require a more stringent review. Region II also sets an upper bound of about 10^{-4} per reactor-year (ry^{-1}) on the baseline CDF. Changes in Region I do not meet the requirements of a small risk increase and will, in general, not be approved.

Fig. 1 also shows gradual shading, darkening as one moves upward and to the right, representing CDF and ΔCDF combinations that are closer to the boundaries between regions. The darkness of the shading corresponds to the level of review that the application will be given, such that LB changes that have a representative point in areas of darker shading, i.e., near the region boundaries, warrant a review that is generally more intensive. "The closer the estimates of ΔCDF and ΔLERF are to their corresponding acceptance guidelines, the more detail will be required" (USNRC, 2002).

RG 1.174 requires that all sources of uncertainty be identified and analyzed such that their impacts are understood at the technical element level, and on the CDF and LERF risk metrics. RG 1.200 states that "an essential aspect of the risk characterization is an understanding of the associated uncertainties (USNRC, 2004a). Uncertainties in the PRA must be understood and accounted for, whether or not they are explicitly modeled.

RG 1.200 also provides two ways to ensure the technical adequacy of a PRA in support of risk-informed regulatory decisions. The first is to meet the criteria of the American Society of Mechanical Engineers (ASME) PRA standard (ASME, 2002), as supplemented by the comments in RG 1.200. The second is to have the PRA peer reviewed using the Nuclear Energy Institute (NEI) peer review process (NEI, 2000), as supplemented by the comments in RG 1.200.

2.1. Types of uncertainty

Uncertainties can be categorized as either aleatory or epistemic uncertainties (Apostolakis, 1993). Aleatory uncertainty reflects our inability to predict random observable events. For example, the flip of a fair coin is generally accepted to yield heads with a probability of 0.50. However, the number of times that heads will occur as a result of 10 coin flips in unknown. Only the probability distribution of the number of heads is known. This is also referred to as 'randomness' or 'stochastic uncertainty.' Epistemic uncertainty represents our confidence in the model and the numerical values of its parameters, e.g., that the coin is fair so that the probability of heads can be taken to be 0.50. A process may not be sufficiently understood and, as such, a specific model may not be universally accepted as being the right model. This type of uncertainty is also called 'state-of-knowledge' uncertainty or just 'uncertainty' (Zio and Apostolakis, 1996). We note that, unlike aleatory uncertainties, epistemic uncertainties are associated with non-observable quantities, e.g., the parameters of models such as failure rates.

The distinction of uncertainty into aleatory and epistemic is largely due to "practical aspects of modeling and obtaining information" (Winkler, 1996). At their core, they both refer to the problem of modeling real-world systems with mathematical formulas, whether deterministic or probabilistic.

Epistemic uncertainties can be roughly split into three categories and is done in RG 1.174. These are: parameter, model, and completeness uncertainty. Parameter uncertainty is that which relates to the parameters of the PRA, given a choice of model. Even with a known model, the parameter values may still be unknown. In situations where historical data is limited, this uncertainty may be quite large. Examples of parameter uncertainties include equipment failure rates, initiating-event frequencies, and human error probabilities.

In many cases, there is limited knowledge and some disagreement on the proper model to represent a system. The result is that for a particular process, there are multiple competing models, each of which necessarily produces a different approximation of the same real-world system. Because the correct model is unknown, there is additional

uncertainty in the output of any model, representing the uncertainty in the model's itself. This uncertainty adds to the parameter uncertainty described above and is model uncertainty. The outputs of each reasonable model must be considered, according to the degree of belief in the appropriateness of each model, to prevent exclusion of valuable uncertainty data from consideration. Several methods have been proposed to accomplish this, including the linear or otherwise combination of models weighted by the analyst's belief that each model may be correct (Apostolakis, 1993), and the use of an adjustment factor on the single most likely model (Zio and Apostolakis, 1996). These and other methods will be discussed in Section 3.

The PRA structure itself is model-dependent because model uncertainty can affect the choice of success criteria. For example, one model might say that two primary relief valves are required to prevent core damage during a loss of offsite power event, while another might say that only one primary relief valve is required. In this case, while there still remains (parameter) uncertainty in the failure rate of relief valves, there is also uncertainty in how many relief valves are required. This latter uncertainty is model uncertainty also. In cases where model uncertainty is treated by using a single, conservative model, the effects of alternate assumptions must be recognized. RG 1.174 recommends using sensitivity studies to determine whether or not there are any assumptions or models whose results would reduce confidence in the conservatism of the chosen model.

Completeness uncertainty is a type of model uncertainty, but is handled differently. It represents the uncertainty due to the portion of risk that is not explicitly included in the PRA. It may be that, for a particular risk contributor, the state-of-the-art has not evolved to the point where the risk can be modeled defensibly. This is the case with safety culture and organizational behavior in general. RG 1.174 states that "the influences of organizational performance cannot now be explicitly assessed." Completeness uncertainty also includes risks that have not been identified. This includes anything that has not been identified as a risk contributor, yet does contribute to risk. Due to the nature of this type of uncertainty, it is impossible to quantify. Conservatisms, such as defense-in-depth and safety margins largely exist to defend against this type of uncertainty, as stated earlier.

Referring back to the acceptance guidelines of Fig. 1, the values of CDF and ΔCDF used are supposed to be epistemic mean values, i.e., the mean values of the distributions of CDF and ΔCDF. These distributions are largely the result of propagating through the PRA the epistemic distributions that represent the parameter uncertainties that were explicitly included in the PRA model. Because these mean values already include the effects of parameter uncertainty, as represented in the probability distributions of the input parameters, they are fairly insensitive to changes in these distributions. In contrast, model uncertainties generally have a greater potential to affect these metrics thus affecting the approval decision. Knudson and Smith support this argument by measuring the model uncertainty regarding the success criteria of Auxiliary Feedwater pumps and comparing it with the parameter uncertainty in the pump failure rates (Knudson et al., 2002). Model uncertainty is measured by varying the number of pumps required, such that the system may be a 1-out-of-3, 2-out-of-3, or 3-out-of-3 system, assigning a probability that each case is true. Looking at each uncertainty while ignoring the effects of the other, the parameter and model uncertainty provide the following 90 percent confidence intervals for the system failure rate.

Parameter uncertainty	2.2×10^{-4} to 6.3×10^{-4} ry^{-1}
Model uncertainty	1.9×10^{-5} to 1.5×10^{-3} ry^{-1}

While the parameter uncertainty range spans about a factor of three, the model uncertainty range spans about two orders of magnitude.

Bley et al. (1992) measure the impact of model uncertainty on CDF directly. In their work on a plant-specific PRA, they identified three model uncertainties that had the greatest potential to impact CDF: reactor coolant pump (RCP) seal LOCA timing, low-end seismic fragility curves for piping and D/C electrical components, and seismically-induced relay chatter. They chose two alternate assumptions for each model and assigned probabilities that each was true. This resulted in eight different sets of assumptions when all three models were inserted into the PRA. Each set of assumptions resulted in a different mean value of CDF for the plant, which ranged from about 2×10^{-4} to 3×10^{-3} ry^{-1}, with the most likely value being about 2×10^{-4} ry^{-1}. The most likely value corresponds to the low end of the range because the set of assumptions with the highest probability of being true resulted in the lowest CDF. These results show that modeling assumptions can shift the mean value of CDF significantly.

3. Model uncertainty

As stated in Section 2.1, there is no mathematical difference between different types of uncertainty. They all refer to unknowns, the limit of knowledge about a real-world phenomenon. "For the case of a finite number of alternative models, the model uncertainty is equivalent to parameter uncertainty" (Buslik, 1993), with reference to Savage's partition problem (Savage, 1972). The theoretical overlap between model and parameter uncertainty can also be seen by creating a parameter whose value is dependent upon the model used (Zio and Apostolakis, 1996).

Methods to deal with model uncertainty include prediction expansion and model set expansion (Zio and Apostolakis, 1996). In prediction expansion, a single model is chosen as the best one to represent the system. However, it is recognized that this model has uncertainties and may model some characteristics of the system better than others. Sensitivity studies are performed on the various assumptions to analyze the effects of the choice of assumptions on the model

output. This uncertainty is dealt with by applying an adjustment factor to the model results. The adjustment factor may be multiplicative or additive, or both may be necessary. The purely additive and multiplicative adjustment factor approaches can be seen in Eqs. (1) and (2), respectively:

$$y = y^* + E_a^* \quad (1)$$

$$y = y^* * E_m^* \quad (2)$$

where y^* represents the model prediction, E^* represents the adjustment factor, and y represents the adjusted model output. E^* may also have (aleatory or epistemic) uncertainty in its value, due to limited data, for example.

In model set expansion, the characteristics of the system under consideration are analyzed and models are created in an attempt to emulate the system based on goodness-of-fit criteria. The models may use different assumptions, and require different inputs. Each model has its own advantages and disadvantages, including limitations on applicability. These models are then combined to produce a meta-model of the system.

Several methods have been proposed regarding the construction of this meta-model. They include mixture (Apostolakis, 1993), Bayesian updating (Winkler, 1993,), the NUREG-1150 approach (Keeney and von Winterfeldt, 1991), the joint US/Commission of European Communities' (EC) Probabilistic Accident Consequence Uncertainty Analysis (PACUA) approach (USNRC, 1997), and the Technical Facilitator-Integrator approach (Budnitz et al., 1996). Of course, all of these methods rely on expert opinion.

In the mixture approach, the set of plausible models is agreed upon from expert opinion and these experts agree on probabilities that each model is correct. The models are then combined linearly, with their weights corresponding to the probability of correctness. The result is a weighted average of the probability distributions that result from each model. The multiple distributions should be presented before they are combined, thus allowing an analyst a more transparent look at the range of models that became the meta-model.

In the Bayesian approach, each model is integrated into the meta-model using Bayes' theorem, using the following formula:

$$f_p(x|f_1, \ldots, f_k) = f(x) * g(f_1, \ldots, f_k|x) \quad (3)$$

where $f_p(x|f_1, \ldots, f_k)$ is the posterior distribution resulting from the combination of the individual models, $f(x)$ is the analysts' prior distribution, $f_i(x)$ is the distribution given by the ith available model, and $g(f_1, \ldots, f_k|x)$ is the likelihood function. This method is theoretically very attractive due its mathematical rigor and ability to incorporate all types of information. However, it is difficult to implement "in large part due to the need to treat the thorny issue of dependence among models" (Winkler, 1993).

In the NUREG-1150 approach, multiple experts are elicited to produce their own probability distribution of the system in question, based on their own opinion. The individual results are then combined linearly, with each expert given equal weight. The PACUA approach goes one step further by including information about the confidence in each expert. The experts are asked to produce distributions for seed variables, for which data is known, and their opinions are compared to the known data. Experts with superior performance when estimating the seed variable distribution are given higher weight when opinions regarding the system in question are elicited.

The final method under consideration here is the technical facilitator-integrator (TFI) approach, which takes advantage of many of the lessons learned from previous expert elicitation exercises. In this approach, the experts are treated as a team, rather than individuals, each sharing their opinion separate from the consideration of the other experts. Individual elicitations are obtained. However, the team works together with the TFI to integrate the data, including the experts' knowledge of technical experts outside of their own group, into a meta-model that attempts to represent the current total body of knowledge. Part of the TFI's role is to mitigate problems identified in behavior science, such as the tendency for more dominant members of the group to be given undue weight on their opinion.

With this background on model uncertainty, the distinction between different classifications of uncertainty, the reason for these classifications, the theory behind the formalisms of model uncertainty, and practical applications, we now look at generic model uncertainties in PRAs that have been identified in the literature.

3.1. Generic model uncertainties

A literature review provided a fairly extensive, yet manageable, list of major model uncertainties pertaining to Level 1, at power, internal events PRAs. Insights from the literature review will be used to learn more about the uncertainties that were identified as important in the case study. Much of the data comes from NRC-sponsored studies. This review was not limited to NRC generated data, however, and a variety of sources was used. A discussion of the results is provided here, with an emphasis on basic events relating to Level 1, at-power, internal events PRAs.

NUREG/CR-4550 (USNRC, 1990) organized an expert panel to address several important uncertainties. Some were related to problems at individual plants and are excluded here. These are:

- Probability of the failure of two check valves in series in a PWR constituting a boundary between a high and a low pressure system.
- Emergency core cooling system (ECCS) failure rates due to venting or containment failure. This refers to the operability of components in hostile environments. PRAs normally assume 100% failure rate if equipment are operated above their qualification limit. This data shows expected failure rates with respect to different types of components, different operating condition, and different lengths of operation.

- RCP seal LOCA probability. Results were given with respect to time after the initiating event and leak rate.
- Probabilities of innovative recovery actions for long-term sequences involving loss of containment heat removal. The panel concluded that success probabilities are highly dependent on plant-specific features like climate, location, staff training, plant design, and layout. Results are given in terms of probabilities of repair versus time for various components.
- Failure probability of using high pressure service water spray in the dry well.
- Battery depletion time.
- Diesel generator field flashing failure probability.
- Hydrogen ignition probability on restoration of A/C power.
- Human actions to shutdown the plant failure probability.
- Secondary safety valve demand and failure rates.
- Reactor coolant system depressurization failure probability.
- Common-cause β-factor uncertainty ranges.
- Common-cause β-factor for air-operated valves.

NUREG-1764 (USNRC, 2004b) provided a list of uncertainties in the reliability of human actions that were either known to be risk-important or had the potential to be risk-important. They are broken into categories by plant type, pressurized water reactor (PWR) or boiling water reactor (BWR), as follows.

Pressurized water reactors:

- Switch the ECCS from the injection mode to the recirculation mode in a LOCA scenario.
- Feed and bleed, particularly the use of pressurizer relief valves.
- Provide water supply for auxiliary feedwater by moving water from alternate sources into the auxiliary feedwater system when long-term cooling is needed.
- Trip the RCPs to prevent RCP seal LOCA on a loss of RCP cooling.
- Recover RCP seal cooling by aligning an alternative means of cooling.
- Recover emergency A/C or offsite power.
- Respond to an anticipated transient without scram (ATWS) – failure of the reactor protection system, particularly the initiation of boron injection and including manual scram of the reactor and ensuring turbine trip.
- Depressurize during a steam generator tube rupture (SGTR). This includes the depressurization of the primary and secondary systems and equalizing pressure between them.
- Isolate steam generator during a main steam leak break or a SGTR.
- Shut power operated relief valve (PORV) blocking valve during a stuck open PORV event.
- Isolate interfacing system LOCA during a LOCA in the low pressure injection system.

Boiling water reactors:

- Perform manual depressurization to allow injection with low pressure injection systems. This is typically done by operating the safety relief valves.
- Vent containment and align containment or suppression pool cooling during a LOCA.
- Control vessel level during an ATWS in order to control reactor power.
- Initiate standby liquid control during an ATWS.
- Inhibit the automatic depressurization system in order to prevent instabilities that occur at low pressures.
- Miscalibration of pressure switches that are important for initiating and controlling the ECCS.
- Initiate isolation condenser in BWR plants of early design.
- Control feedwater events. Control the feedwater system after a loss of feedwater event.
- Recover offsite power.
- Shed D/C loads after a Station Blackout in order to extend battery life.

Bley, Buttemer, and Stetkar argue that an adequate analysis of event sequence timing is important in PRA analysis for a couple of reasons (Bley et al., 1988). Success criteria determination requires an understanding of sequence timing. How plant parameters change over time during an accident sequence determines what equipment is necessary to prevent core damage. Success criteria are often chosen based on deterministic thermohydraulic calculations using assumptions that are conservative. Calculation of the probability of recovery is also dependent on the results of a sequence timing analysis. Human performance is highly dependent on the time available for the operator to complete actions, and the time available is calculated using an analysis of sequence timing. The authors analyze a number of risk-important parameter and model uncertainties and reach the following conclusions: success criteria and recovery modeling are highly dependent on sequence timing; determinations of the factors that affect operator performance, including dependencies and competing demands, requires a detailed analysis; and simple analysis involving mass and energy balance to determine their effect on sequence timing is often sufficient for PRA applications.

RG 1.200 (USNRC, 2004a) provides several examples of key uncertainties when determining the technical adequacy of a PRA. Uncertainties in success criteria, human reliability, and the choice of a RCP seal LOCA model are included. In these cases, the choice of the model and how it is used may have a significant impact on risk.

Sump performance was identified as important by the NRC's Advisory Committee on Reactor Safeguards (Wallis, 2004). Because of the nature of the sump, and the limited data that exist on how a sump might perform when needed, it is difficult to estimate the probability that it will perform successfully. Specifically, it is difficult to model how the strainer on the intake side of the pump will be affected by debris in

the sump. There is some probability that the debris will clog the strainer and reduce the net positive suction head on the pump sufficiently to effectively disable the pump.

Interviews with NRC personnel also provided a number of important model uncertainties (Interview). RCP seal LOCA probability, battery depletion time, common-cause failure modeling, and modeling of sump plugging and pool strainer plugging were identified. Emergency diesel generator mission time and recovery modeling were identified also. This refers to how long the diesel generator is assumed to be needed in order to fulfill its mission, and also what mechanisms for recovery are credited in the PRA and the probability of these recovery mechanisms. Success criteria determination is important, specifically with regard to how many PORVs are required during a feed-and-bleed evolution. Support systems may be important in ways that are not obvious at first glance, especially when they have the ability to cause common-cause failures across many systems. Sometimes, the risk-importance of these systems is missed and they are either not modeled adequately, introducing model uncertainty, or left out of the PRA, introducing completeness uncertainty. Instrument air is an example of a support system that many components in multiple systems depend on, but that may not seem risk-important itself unless attention is brought to these dependencies. The modeling of SGTR event tree was also considered important.

4. Proposed methodology

The CDF is calculated as

$$\mathrm{CDF}_{\mathrm{base}} = \sum_i \mathrm{fr}(\mathrm{MCS}_{\mathrm{base},i}) \tag{4}$$

where $\mathrm{CDF}_{\mathrm{base}}$ is the baseline CDF of the plant, as it is normally configured. However, this calculation can be performed for any plant configuration. $\mathrm{MCS}_{\mathrm{base},i}$ is the ith minimal cut set, and $\mathrm{fr}(\mathrm{MCS}_{\mathrm{base},i})$ is the frequency at which the ith cut set occurs in the baseline PRA model.

Uncertainties surround the value of the baseline CDF, since there are uncertainties in the frequency of the initiating events and also in the conditional probabilities of occurrence of the basic events. These same uncertainties create uncertainties in the value of ΔCDF. The uncertainties in ΔCDF can have a significant impact on the decision whether or not to approve the change, as acceptance guidelines are provided as a combination of CDF and ΔCDF. The significance of this impact can be seen by looking at the definition of ΔCDF,

$$\Delta\mathrm{CDF} = \mathrm{CDF}_{\mathrm{after}} - \mathrm{CDF}_{\mathrm{base}}, \tag{5}$$

where $\mathrm{CDF}_{\mathrm{after}}$ is the CDF of the plant after the proposed licensing change. Inserting Eq. (4) into Eq. (5), we find that

$$\Delta\mathrm{CDF} = \sum_i \mathrm{fr}(\mathrm{MCS}_{\mathrm{after},i}) - \sum_j \mathrm{fr}(\mathrm{MCS}_{\mathrm{base},j}) \tag{6}$$

where $\mathrm{MCS}_{\mathrm{after},j}$ is the frequency at which the jth cut set occurs in the PRA model as it exists after the proposed licensing basis change. The proposed change will change the frequency at which some of the minimal cut sets occur. However, most of them will remain unchanged. Each minimal cut set that is unaffected will, therefore, appear in both terms on the right-hand side of Eq. (6) and drop out of the equation. It is clear that uncertainties affect the value of CDF and ΔCDF. It is also clear that these uncertainties can change the outcome of a decision based on the acceptance guidelines in Fig. 1.

We propose a methodology for including these uncertainties in the decision-making process used to make risk-informed licensing basis decisions in accordance with RG 1.174. This methodology begins with using the PRA of a plant to determine the RAW importance measure of each basic event.

4.1. RAW with respect to CDF

Importance measures are used in the ranking and categorization of basic events modeled in a PRA (Cheok et al., 1998). The importance measure of most interest to us is the risk achievement worth (RAW). RAW is defined as

$$\mathrm{RAW}_j = \frac{R_j^+}{R} \tag{7}$$

where RAW_j is the value of RAW for basic event j, R is the value of the model's baseline risk metric, and R_j^+ is the value of the model's risk when basic event j is set to a logical TRUE. Each basic event, therefore, is assigned a value of RAW, by the PRA, that quantifies the factor by which a plant's risk would increase if the associated basic event were assumed to be completely unreliable. RAW is a bounding measure that provides the maximum level of risk that a basic event could cause (Cheok et al., 1998).

The meaning of RAW can also be viewed with respect to the logic structure of the PRA. A basic event that is completely unreliable serves no risk function in the PRA. It is as if the basic event were completely removed from the logic structure. Therefore, the RAW of a basic event represents the factor by which a plant's risk would increase if the basic event were removed from the plant. Since the risk metric in this case is the CDF,

$$\mathrm{RAW}_{j,\mathrm{CDF\text{-}base}} = \frac{\mathrm{CDF}_{j,\mathrm{base}}^+}{\mathrm{CDF}_{\mathrm{base}}}. \tag{8}$$

The set of values for $\mathrm{RAW}_{j,\mathrm{CDF\text{-}base}}$ can easily be generated using the SAPHIRE program.

4.2. RAW with respect to ΔCDF

Importance measures can also be used to show areas in a PRA where uncertainty can have the greatest impact on the change in risk that is proposed by the licensing basis change. To represent this importance measure, we start with the definition of RAW above, Eq. (7), and note that the risk metric R in this case is ΔCDF. Therefore

$$\mathrm{RAW}_{j,\Delta\mathrm{CDF}} = \frac{\Delta\mathrm{CDF}_j^+}{\Delta\mathrm{CDF}} \tag{9}$$

Noting the definition of ΔCDF, Eq. (5), we expand this equation to

$$\mathrm{RAW}_{j,\Delta\mathrm{CDF}} = \frac{\mathrm{CDF}_{j,\mathrm{after}}^+ - \mathrm{CDF}_{j,\mathrm{base}}^+}{\mathrm{CDF}_{\mathrm{after}} - \mathrm{CDF}_{\mathrm{base}}} \tag{10}$$

Inserting Eq. (8) into Eq. (10), we see that

$$\mathrm{RAW}_{j,\Delta\mathrm{CDF}} = \frac{(\mathrm{RAW}_{j,\mathrm{CDF\text{-}after}}) * (\mathrm{CDF}_{\mathrm{after}}) - (\mathrm{RAW}_{j,\mathrm{CDF\text{-}base}}) * (\mathrm{CDF}_{\mathrm{base}})}{\mathrm{CDF}_{\mathrm{after}} - \mathrm{CDF}_{\mathrm{base}}} \tag{11}$$

The values on the right-hand side of Eq. (11) are fairly easy to generate. From the PRA, $\mathrm{CDF}_{\mathrm{base}}$ is known directly. From the application of Eq. (8), we calculate the set of values for $\mathrm{RAW}_{j,\mathrm{CDF\text{-}base}}$. To find the other values, we must modify the PRA to represent the plant as it would exist after the change. Using this model and repeating the steps used to calculate $\mathrm{CDF}_{\mathrm{base}}$ and $\mathrm{RAW}_{j,\mathrm{CDF\text{-}base}}$, we calculate $\mathrm{CDF}_{\mathrm{after}}$ and the set of values for $\mathrm{RAW}_{j,\mathrm{CDF\text{-}after}}$. Now, all of the variables on the right-hand side of this equation are known and the set of values for $\mathrm{RAW}_{j,\Delta\mathrm{CDF}}$ can be generated using, for example, sorting and arithmetic algorithms in a Microsoft Excel spreadsheet.

4.3. Calculating RAW thresholds

At this point, we have a complete set of basic events with their respective values for RAW with respect to CDF and RAW with respect to ΔCDF. Some threshold must be set to determine the value of RAW below which we deem the basic event to be not risk-important. It should be noted that traditionally in licensing basis change requests a threshold value of RAW is set at a value of two (NEI, 1996). Using this method, basic events with a RAW of two or higher are deemed as potentially important and analyzed further, while those with a RAW less than two are classified as not risk-important. The methodology in this paper proposes a simple method for determining a change-specific threshold value of RAW.

By referring to the acceptance guidelines in Fig. 1 and the position of the proposed licensing change's risk on the figure, we see that there is some value of RAW that will move the plant's risk to the right on the figure until it enters a different region. The RAW with respect to CDF of each basic event indicates the maximum amount that uncertainty in this basic event can move the plant's CDF to the right. A small RAW might indicate that regardless of the reliability of a particular basic event, the CDF would not be in Region I, and therefore would not affect the decision. In other cases, the RAW may be large enough. It is only when the CDF moves into Region I that the uncertainty becomes important to the decision. We, therefore, determine the threshold value of RAW with respect to CDF that will change the decision as follows:

$$\mathrm{RAW}_{\mathrm{CDF,threshold}} = \frac{\mathrm{CDF}_{\mathrm{threshold}}}{\mathrm{CDF}_{\mathrm{base}}} \tag{12}$$

where $\mathrm{RAW}_{\mathrm{CDF,threshold}}$ is the RAW value that will move the CDF to the right and into a different region, and $\mathrm{CDF}_{\mathrm{threshold}}$ is the value of CDF corresponding to the vertical lines between the regions of Fig. 1. The threshold RAW value is dependent on the CDF of the plant and the ΔCDF of the proposed change and its value is change-specific. Remember that although the CDF has uncertainty and is represented by a distribution of values, RG 1.174 calls for the mean value to be used in Figs. 1 and 2.

The threshold value for ΔCDF can be determined in a similar fashion. Referring to Fig. 1, we see that there is a value of RAW with respect to the ΔCDF that will move ΔCDF upward in the figure until it enters a different region. It is this change between regions that changes the context of the decision and possibly the decision itself. We, therefore, determine the value of RAW with respect to ΔCDF that will change the decision as follows:

$$\mathrm{RAW}_{\Delta\mathrm{CDF,threshold}} = \frac{\Delta\mathrm{CDF}_{\mathrm{threshold}}}{\Delta\mathrm{CDF}} \tag{13}$$

where $\mathrm{RAW}_{\Delta\mathrm{CDF,threshold}}$ is the RAW value that will move ΔCDF upward and into a different region, and $\Delta\mathrm{CDF}_{\mathrm{threshold}}$ is the value of ΔCDF corresponding to the horizontal line between the applicable regions of Fig. 1. Just as the $\mathrm{CDF}_{\mathrm{base}}$ used when calculating $\mathrm{RAW}_{\mathrm{CDF,threshold}}$ was a mean value, the ΔCDF value used here should be a mean value. This determination again differs from the traditional RAW threshold value of two used in licensing basis change decisions. The threshold value of RAW used here is change-specific.

4.4. Cross-check of important basic events

Of course, not all basic events have large uncertainties in their reliabilities. For example, motor-driven pumps have been used extensively in nuclear power plants for some time. Because of this, a sufficient amount of historical failure data has been accumulated such that their failure rates are known with a fair degree of certainty and the mechanisms by which they fail are fairly well understood. The methodology, therefore, cross-checks the basic events whose uncertainty may be important as identified by the method above to basic events that have been identified in the literature review as having generally important model uncertainty. The basic events that remain after the cross-check are those that are important to the plant-specific licensing basis change decision at hand. They have model uncertainties identified in the literature as generically important, and are close enough to a threshold value in Fig. 1 that these uncertainties could possibly affect the decision. The generically important model uncertainties and their descriptions are included in Section 3.

4.5. Making the decision

Having identified the important basic events with respect to both CDF and ΔCDF, we must now investigate their potential impact on the decision. Quantifying the model uncertainty would allow the decision maker to see how CDF and ΔCDF of the proposed change move and whether the change meets the acceptance guidelines. However, model uncertainty is quite difficult to quantify at this time. Instead, our proposed methodology employs sensitivity analysis to determine the degree to which a basic event's failure probability would need to change in order to violate the acceptance guidelines, in effect changing the approval decision. From this point, qualitative arguments remove some basic events from further consideration. This does, of course, still rely on expert opinion as a tool for estimating plausible upper bounds for risk. However, expert opinion is used to a lesser degree because many uncertainties are eliminated from consideration because they are determined to be unimportant. Remaining basic events must be subjected to considerable scrutiny and the effects of uncertainty quantified before a decision can be made. Calculations needed for this paper were performed using the Standardized Plant Analysis Risk (SPAR) model for this plant and the System Analysis Programs for Hands-On Integrated Reliability Evaluations (SAPHIRE) computer software. Model uncertainties that require a detailed quantitative evaluation may be handled by using the methods of Section 3. For example, one method was to use an adjustment factor. In this case, reasonable alternative assumptions to those used in the PRA are established, and their effects on the PRA output quantified. Expert judgment is then used to determine the probability distribution of the adjustment factor. The adjusted CDF from the PRA is then compared with the acceptance guidelines to determine if the licensing basis change decision is sensitive to these modeling assumptions.

The benefit of the proposed methodology is that important basic events are identified with respect to the change-specific decision at hand rather than using a general importance measure. This methodology also reduces the number of uncertainties that must be analyzed extensively by allowing qualitative arguments based on change-specific conditions. In its present form, however, this methodology is not sufficient when model uncertainty affects the logical structure of the PRA, as in the example provided earlier where uncertainty affected the success criteria.

5. The case study

To illustrate the methodology proposed in this paper, we present a case study. This case study was a licensing basis change request submitted to the NRC. The request applied to a commercial PWR Westinghouse four-loop design and proposed to establish a risk-informed in-service testing (IST) program that would replace the existing IST requirements for a portion of the plant's valves. The IST program applied to 160 valves that made up various portions of 10 systems. These systems were:

steam generator blowdown;
heating and ventilation – purge air;
compressed air – control air;
chemical and volume control;
safety injection;
essential raw cooling water;
component cooling;
core spray;
waste disposal;
radiation monitoring.

The risk-informed IST program proposed that, for these valves, the IST frequency would be changed from once per quarter to once per refueling cycle, or approximately once per 18 months. The licensee states that it is conservative to assume that the failure probability of these valves increases linearly with time between inspections. Since the time between inspections increases by about a factor of six, the failure probability of each valve affected by the change is increased by a factor of six when modeling the effects of the change.

The point estimate baseline CDF and ΔCDF of the plant were:

$$\mathrm{CDF} = 6.8 \times 10^{-5}\ \mathrm{ry}^{-1}$$

$$\Delta\mathrm{CDF} = 6.9 \times 10^{-7}\ \mathrm{ry}^{-1}$$

These values represent a point in the CDF versus ΔCDF acceptance guidelines as shown in Fig. 3. This pair of values places the point representing the proposed LB change in Region III of the acceptance guidelines. We note that the CDF and ΔCDF reported were "point" estimates, i.e., the licensee did not propagate the distributions of the input parameters to produce distributions for CDF and ΔCDF. As point estimates, the values of CDF and ΔCDF are sensitive to parameter uncertainties also. Since our objective is to investigate model uncertainties, we will treat these point estimates as if they were epistemic means.

Following our methodology, we must first generate the complete set of RAW values for the basic events at this plant. This includes the RAW with respect to CDF and the RAW with respect to ΔCDF.

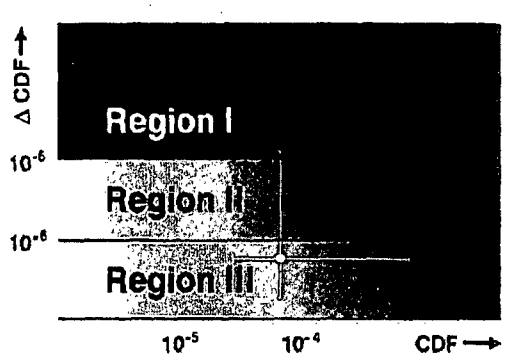

Fig. 3. CDF acceptance guidelines with representative point.

5.1. Event RAW with respect to CDF

The point representing the proposed change is in Region III and the decision would be affected if uncertainties moved this point into Region I. Therefore, we are interested in the horizontal threshold between Region I and Region III, which is about $10^{-3}\,ry^{-1}$. Although this is not a "bright line" boundary, $10^{-3}\,ry^{-1}$ is a reasonable value. We note that the NRC has a goal of keeping the CDF below $10^{-4}\,ry^{-1}$, thus making $10^{-4}\,ry^{-1}$ another reasonable boundary value; this is also the value it would have to exceed to be in Region I if the uncertainties were to bring the ΔCDF up to Region II. Therefore, $CDF_{threshold}$ is given two values, equal to 10^{-3} and $10^{-4}\,ry^{-1}$. The CDF_{mean} of the plant was $6.8\times10^{-5}\,ry^{-1}$. Therefore, the $RAW_{CDF,threshold}$ values were about 14.6 and 1.46, respectively, which we truncated to 14 and 1.4. Any basic event with a RAW greater than 14 has the potential to change the licensing basis decision because the actual CDF could be in Region I. Also, a RAW greater than 1.4 indicates that uncertainty in the basic event probability is important to the decision because the actual CDF could be greater than the NRC goal of keeping CDF less than 10^{-4}, and it has the potential of being important to the decision, depending on the ΔCDF value once uncertainties are included.

Using SAPHIRE, we determined that there were 12 basic events with RAW greater than 14. They are listed in Table 1.

We also found that there were 32 additional basic events with RAW greater than 1.4. They are listed in Table 2.

5.2. Event RAW with respect to ΔCDF

In our case study, the decision would be affected if model uncertainties moved the representative point into Region I or Region II; so we are interested in the vertical boundary between Region III and Region II at about $10^{-6}\,ry^{-1}$ and between Region II and Region I at about $10^{-5}\,ry^{-1}$. ΔCDF_{mean} was given in the licensee's application as $6.9\times10^{-7}\,ry^{-1}$. Our thresholds, $\Delta CDF_{threshold}$ of $10^{-6}\,ry^{-1}$ and $\Delta CDF_{threshold}$ of $10^{-5}\,ry^{-1}$, yield $RAW_{\Delta CDF,threshold}$ values of 1.4 and 14, respectively. Therefore, any basic event with a RAW with respect to the ΔCDF greater than 1.4 has the potential to change the decision. If this RAW is 14, then the potential to change the decision is much higher because the representative point would be in Region I.

Using SAPHIRE, we determined that there were 10 basic events that had a RAW with respect to ΔCDF greater than 14. They are listed in Table 3.

For each of these basic events, the RAW with respect to CDF, as calculated in Section 4.1, is provided for comparison. Out of these 10 events deemed important to the licensing basis decision because of their RAW with respect to ΔCDF, only three were identified as important because their RAW with respect to CDF exceeded 14. These are emphasized with bold font in Table 3. This implies that

Table 1
RAW with respect to CDF: RAW > 14

Basic event	RAW_{CDF}
(a) Control rods fail to insert	3050
(b) Common-cause diesel generator failure	271
(c) Failure to depressurize due to hardware failure	218
(d) Scram breakers fail to open	202
(e) 4160V Bus 1B fails	197
(f) Common-cause failure of residual heat removal (RHR) pumps	134
(g) Common-cause failure of RHR heat exchangers	134
(h) Reserve water storage tank not available	112
(i) Common-cause auxiliary feedwater pump failure	26.6
(j) Common-cause failure of steam generator discharge valves to open	26.5
(k) Common-cause failure of the steam generator inlet check valves	26.0
(l) 4160V Bus 1A fails	18.4

Table 2
RAW with respect to CDF: 1.4 < RAW < 14

Basic event	RAW_{CDF}
(a) Diesel generator B fails	12.1
(b) Control rods remain energized	11.0
(c) Diesel generator A fails	11.0
(d) Operator failure to depress below steam generator relief valve setpoints	8.89
(e) Failure to recover offsite power before battery depletion	4.83
(f) Failure to isolate faulty steam generator	4.26
(g) Ruptured steam generator isolations fail	4.23
(h) Turbine boundary valves and condenser fail to cooldown the reactor coolant system	3.75
(i) Common-cause failure of RHR suction valves	3.66
(j) Operator fails to initiate RHR	3.66
(k) Operator fails to isolate reserve water storage tank	3.65
(l) RHR hotleg discharge valve A fails	3.65
(m) RHR hotleg discharge valve B fails	3.65
(n) PORV 1 fails to reclose	3.52
(o) Operator failure to initiate cooldown below RHR tolerances	3.40
(p) RCP seals fail	3.30
(q) Common-cause failure of RHR to both high pressure injection isolation valves	2.93
(r) Common-cause failure of sump recirculation valves	2.93
(s) Common-cause failure of reserve water storage tank isolations	2.93
(t) Sump failure	2.92
(u) Operator failure to initiate high pressure recirculation	2.89
(v) PORV 1 fails to open	1.96
(w) Common-cause failure of auxiliary feedwater motor-driven pumps	1.87
(x) Auxiliary feedwater steam supply valves fail	1.86
(y) Common-cause failure of turbine driven pump steam supply valves to open	1.85
(z) Turbine driven pump	1.84
(aa) Operator fails to identify a SGTR	1.77
(ab) Operator fails to initiate feed and bleed	1.77
(ac) Operator fails to initiate reactor coolant system depressurization	1.76
(ad) RHR motor driven pump B fails	1.59
(ae) RHR motor driven pump A fails	1.53
(af) Operator fails to manually scram reactor	1.43

Table 3
RAW with respect to ΔCDF: RAW > 14

Basic event	$RAW_{\Delta CDF}$	RAW_{CDF}
(a) Failure to isolate faulty steam generator	55.5	4.26
(b) Mechanical failure of steam generator isolations	55.0	4.23
(c) 4160V Bus 1B fails	**44.2**	**197**
(d) Common-cause failure of RHR pumps	**35.6**	**134**
(e) Failure to initiate high pressure recirculation	33.6	2.89
(f) Common-cause failure of RHR supply to high pressure injection isolation valves	33.6	2.94
(g) Common-cause failure of sump recirculation valves	33.6	2.94
(h) Common-cause failure of RHR reserve water storage tank isolation valves	33.6	2.94
(i) Sump failure	33.4	2.93
(j) Common-cause failure of RHR heat exchangers	**32.8**	**134**

Table 4
RAW with respect to ΔCDF: 1.4 < RAW < 14

Basic event	$RAW_{\Delta CDF}$
(a) Common-cause failures of high pressure injection flowpath	4.45
(b) High pressure injection cold leg injection valve fails	4.38
(c) Common cause failure of reactor coolant system cold leg discharge check valves	4.38
(d) High pressure injection serial component failures	4.23
(e) Common-cause failure of high pressure injection discharge check valves	4.16
(f) 4160V Bus 1A fails	2.79
(g) Reserve water storage tank not available	2.35
(h) Scram breakers fail to open	2.34
(i) Operator fails to diagnose SGTR	1.94
(j) Operator fails to initiate depressurization	1.93
(k) Operator fails to throttle high pressure injection to reduce pressure	1.91
(l) Common-cause failure of chemical & volume control discharge valves A and B	1.70
(m) Common-cause failure of chemical & volume control discharge valves C and D	1.70
(n) Charging system discharge check valves fail	1.70
(o) Charging system suction check valves fail	1.70
(p) Common-cause failure of charging pumps	1.70
(q) Common-cause failure of chemical & volume control suction valves	1.61
(r) Common-cause failure of VCT isolation valves	1.61
(s) Common-cause failure of chemical & volume control pump check valves	1.60
(t) RHR motor driven pump B fails	1.45
(u) RHR discharge valve B fails	1.43
(v) Sump isolation valve B fails	1.43
(w) Reserve water storage tank isolation valve B fails	1.43
(x) RHR discharge A fails	1.43
(y) Sump isolation valve A fails	1.43
(z) Reserve water storage tank isolation valve A fails	1.43
(aa) RHR motor driven pump A fails	1.42
(ab) Fail to depressurize due to hardware	1.40

Table 5
RAW with respect to CDF and RAW with respect to ΔCDF > 14

Basic events	$RAW_{\Delta CDF}$	RAW_{CDF}
(a) 4160V Bus 1B fails	44.2	197
(b) Common-cause failure of RHR pumps	35.6	134
(c) Common-cause failure of RHR heat exchangers	32.8	134

Table 6
RAW with respect to CDF > 14 and RAW with respect to ΔCDF > 1.4

Basic events	$RAW_{\Delta CDF}$	RAW_{CDF}
(a) Operator failure to isolate a faulty steam generator	55.5	4.26
(b) Ruptured steam generator isolation failures	55.0	4.23
(c) Operator fails to initiate high pressure recirculation	33.6	2.89
(d) Common-cause failure of RHR supply to high pressure injection valves	33.6	2.94
(e) Common-cause failure of sump recirculation valves	33.6	2.94
(f) Common-cause failure of residual heat removal reserve waster storage tank isolation valves	33.6	2.94
(g) Sump failure	33.4	2.93
(h) 4160V Bus 1A fails	2.79	18.4
(i) Reserve water storage tank not available	2.35	112
(j) Scram breakers fail to open	2.34	202
(k) Failure to depressurize due to hardware failure	1.40	218
(l) Operator fails to diagnose SGTR	1.94	1.77
(m) Operator fails to initiate depressurization	1.93	1.76
(n) Operator fails to throttle high pressure injection to reduce pressure	1.91	1.77
(o) RHR motor driven pump 1B fails	1.45	1.60
(p) RHR motor driven pump 1A fails	1.42	1.53

although, individually, uncertainty in the remaining seven basic events cannot be sufficient to move the licensee's CDF horizontally into Region I of the acceptance guidelines, they each have uncertainty that may be sufficient to move the change vertically into Region 1.

We also found that there were 28 additional basic events that had RAW with respect to ΔCDF greater than 1.4. They are listed in Table 4.

5.3. Combined importance with respect to CDF and ΔCDF

Basic events that have high RAW values with respect to both CDF and ΔCDF are especially important because their uncertainty can move the representative point both horizontally and vertically in Fig. 1 simultaneously. In our case study, a factor of 14 increase in CDF or a factor of 14 increase in ΔCDF was sufficient to move the point into Region I. However, a factor of 1.4 increase in CDF in combination with a factor of 1.4 increase in ΔCDF

would also move the representative point into Region I. This is because the ΔCDF required to enter Region I changes, depending on the value of CDF. If CDF is greater than about $10^{-4}\,\mathrm{ry}^{-1}$, then ΔCDF must be below about $10^{-6}\,\mathrm{ry}^{-1}$ to remain out of Region I. Otherwise, ΔCDF may be as large as $10^{-5}\,\mathrm{ry}^{-1}$. This threshold RAW value of 1.4 is an order of magnitude lower than the previously required threshold RAW values of 14. Therefore, basic events with uncertainties that affect both the CDF and ΔCDF, but have a relatively weak effect on each, can still affect the licensing basis decision. The fact that the factor of 14 increase required is the same for both CDF and ΔCDF is purely coincidental.

For basic events that were important with respect to both CDF and ΔCDF, we divided them into three categories. There were three basic events that had both a RAW with respect to CDF and a RAW with respect to ΔCDF greater than 14. They are listed in Table 5.

In addition to those listed in Table 5, there were 11 basic events that had both a RAW with respect to CDF and a RAW with respect to ΔCDF greater than 1.4. They are listed in Table 6.

6. Evaluation

Thus far, we have identified basic events whose probability has the potential to adversely affect the decision. Next, it must be determined whether there are model uncertainties associated with these basic events that can actually lead to a different decision. We start by matching the generically important model uncertainties from the literature with the basic events identified in the tables and then determine how far the probability of each basic event would have to shift in order to impact the decision. Expert opinion must be used to determine whether or not the required shift is reasonable. We provide some analysis here, but this is an area that requires further research, perhaps building upon the ideas presented in Section 3.

The basic events in Table 1 had high RAW with respect to CDF, sufficient to move the representative point from the case study horizontally into Region I. Table 1 was only concerned with the effects of uncertainty on CDF, without regard to their effect on ΔCDF. Basic events (b), (f), (g), (i), (j), and (k) in Table 1 are all similar in that they refer to a common-cause failure mechanism. Basic event (b) in Table 1, "Common-cause diesel generator failure", is particularly interesting because it ranks second in importance with respect to CDF and is described in the literature as generally important. Specifically, the modeling of diesel generator mission time and recovery are important model uncertainties (Interview). These uncertainties are related to how long the diesel is assumed to be needed in order to accomplish its mission, and how probabilities of recovering a failed diesel generator are calculated. In the model used for this analysis, a single bounding mission time of four hours was chosen and diesel generator failure probabilities were calculated using this value. Therefore, if an event required a mission time shorter than four hours, the calculated failure probability would be conservative. If an event required longer than four hours of operation, the failure probability would be optimistic. The mission time must therefore be chosen to bound reasonable mission time requirements in order to maintain conservatism. However, the reasonableness of this bound may change if, for example, confidence in the reliability of the municipal electric grid changes.

Modeling of diesel generator field flashing success probabilities have also been identified as generally important to diesel generator failure rates (USNRC, 1990). Because of diesel generator design, field flashing is a necessary component of a generator's ability to produce electricity and, therefore, has a large impact on diesel generator failure rates. During a station blackout, where offsite power and emergency A/C power have been lost, diesel generator field flashing power is drawn from station batteries. Therefore, the duration the battery is capable of supplying power before it is depleted is also a factor in determining diesel generator failure rates. Battery depletion time has also been identified as an important model uncertainty (USNRC, 1990).

Sensitivity to these model uncertainties can be found by varying the specific assumptions related to each. For example, the modeling of diesel generator field flashing is important. One could look at the model used to determine the probability that field flashing would fail, and question the assumptions that it makes. In lieu of this, we vary the failure rate of the basic event, effectively using the failure rate as a proxy for the modeling assumptions that went into its determination.

The basic event that we are concerned with here is "Common-cause diesel generator failure". Since there are two diesel generators, the failure rate of this event is calculated as follows:

$$\lambda_c = \beta(\lambda_{fts} + \lambda_{ftr} * t) \tag{14}$$

where λ_c is the total common-cause failure rate, λ_{fts} is the rate of independent diesel generator failure to start on demand, λ_{ftr} is the rate of independent diesel generator failure while running per hour, t is the mission time in hours, and β is the beta-factor, defined as

$$\beta = \frac{\lambda_c}{\lambda_i + \lambda_c} \tag{15}$$

where λ_c is the common-cause failure rate and λ_i is the independent failure rate (represented by the sum of λ_{fts} and $\lambda_{ftr} \times t$). Looking at Eq. (14), we see that the common-cause failure rate has two components, β and the independent failure rate. In the PRA, β is 0.038, λ_{fts} is 3.0×10^{-2}/demand, λ_{ftr} is 2.0×10^{-3}/hour and the mission time is four hours. The common-cause failure rate is, therefore,

$$\begin{aligned}\lambda_c &= 0.038 \times [3.0 \times 10^{-2}/\text{demand} + (2.0 \\ &\quad \times 10^{-3}/\text{hour})(4 \text{ hour mission time})] \\ &= 1.44 \times 10^{-3} \text{ per mission.}\end{aligned} \tag{16}$$

We tested the sensitivity of the CDF to variations in λ_c and found that a factor of 35 increase would change the common-cause failure rate to 0.051 and place the representative point in Region I of the acceptance guidelines. The question now is whether this increase is reasonable. Since λ_c is a product of two variables, we must question the values used for each variable in order to question to value used for λ_c.

We first look at the value of β. In the PRA, β is 0.038. The value of 0.10 is often used as a generic value for β. The NRC's Common-Cause Failure Database (CCFDB) (USNRC, 2001) provides common-cause failure data from industry-wide operational experience. It provides failure-while-running and failure-to-start data for diesel generators. In this database, the value of β for the failure-while-running case has a mean of 0.0370 and a 95th percentile of 0.0499. β for the failure-to-start has a mean of 0.0263 and a 95th percentile of 0.0370. These are lower than the generic value of 0.10, indicating that diesel generators are somewhat robust with regard to common-cause failures. This is due to the fact that diesel generators are well known to be risk-important and focused efforts have been made to minimize the fraction of common-cause failures. Notably, the Station Blackout rule, 10 CFR 50.63, established the emergency diesel generator reliability program. The value of β in the PRA is consistent with the CCFDB values.

We next look at the value of λ_i, the independent diesel generator failure rate. In our case study, λ_i is quantified as

$$\begin{aligned}\lambda_i &= 3.0 \times 10^{-2}/\text{demand} + (2.0 \times 10^{-3}/\text{hour}) \\ &\quad \times (4 \text{ hour mission time}) \\ &= 0.038 \text{ per mission} \end{aligned} \tag{17}$$

The failure rate is a function of the probability that the diesel generator fails to start and the probability that it fails to run for the mission time. We compared the probabilities used in the case study with the probabilities used in representative PWR PRAs and found them to be consistent. However, these sources used the same NRC Accident Sequence Evaluation Program (ASEP) database, which uses industry-wide accumulated data. Plant-specific failure rates may vary considerably from the industry averages. Also, the electrical grid outage of August 14, 2003 has raised issues as to whether the current modeling assumptions are sufficient (Rasmusson, 2004), specifically, assumptions on the time to recover offsite power. Because of the extent of this outage, recovery times at some plant were quite long, raising concern that current recovery times that are modeled may not be long enough. Therefore, the basic event failure probability may be optimistic.

The important model uncertainties were diesel generator mission time and recovery modeling, diesel generator field flashing modeling, and battery depletion time modeling. Mission time modeling assumptions may be optimistic in our case study because of the recent question of whether mission times are long enough. Diesel generator field flashing and battery depletion time modeling may be optimistic in our case study for the same reasons. Longer outages require longer battery depletion times in order to prevent a station blackout. Also, the battery is required to supply diesel generator field flashing power.

We have concluded that a β value of 0.037 is reasonable. So, in order for the common-cause failure rate to be 0.051 (sufficient to affect the decision), the independent failure rate would have to increase to

$$\lambda_i = \frac{\lambda_c}{\beta} = \frac{0.051}{0.038} = 1.34 \text{ per mission} \tag{18}$$

This value is clearly unrealistic, therefore, model uncertainties regarding the diesel generators do not appear to be capable of affecting the decision.

Looking at the basic events that were important with respect to ΔCDF in Table 3, we see that three of the uncertainties involve known important model uncertainties. They are listed in Table 7 with their associated model uncertainties.

Basic event (a) in Table 3, "Failure to isolate faulty Steam Generator" is recognized as having the potential to be a risk-important human action after a Main Steam leak or a Steam Generator Tube Rupture initiating event. In the PRA, the conditional probability that this action will not be done when needed is 10^{-3}. For this analysis, we must increase this failure probability by a factor of 250 (thus making it 0.25) to achieve a RAW with respect to the ΔCDF of 14, thus placing the representative point in Region I. This same factor of 250 increases the CDF by only a factor of two. It is apparent, therefore, that the effect of an uncertainty on the CDF and the ΔCDF can be quite different.

Uncertainty in human reliability is well known to be important. Inputs to human reliability models, such as performance shaping factors, are difficult to quantify, the models are sensitive to these inputs, and different human reliability models with the same inputs may produce failure rates that span orders of magnitude. In the European Commission's Human Factors Reliability Benchmark Exercise (Poucet, 1989), 15 teams of analysts from different countries were asked to calculate human reliability for the

Table 7
Associating basic events with model uncertainties

Basic event	$RAW_{\Delta CDF}$	Associated model uncertainty
Failure to isolate faulty steam generator	55.5	Human reliability – failure to isolate faulty steam generator (USNRC, 2004b)
Failure to initiate high pressure recirculation	33.6	Human reliability – switch ECCS from injection to recirculate (USNRC, 2004b)
Failure of sump	33.4	Sump plugging and pool strainer plugging modeling (Interview)

crew's response to an operational transient at a nuclear power plant. One team produced results using different models ranged from about 1.5×10^{-2} to 3.5×10^{-1}. Across teams, results using the same model ranged from about 6×10^{-3} to 3.5×10^{-1}. In order for the "Failure to isolate faulty Steam Generator" basic event to be important to the decision, the probability of not performing this action would need to change from one in 1000 to one in four. To assess whether or not an error probability of 0.25 is reasonable for this event, one would need to look at operator training, time available, and other performance shaping factors. Such a high probability, however, does appear to be unreasonable.

Basic event (e) of Table 3, "Failure to initiate High Pressure Recirculation" is another human action and has the potential to be important to the decision. In the PRA, the conditional probability that this action will not be done when needed is 10^{-3}. For the analysis, we increased this failure probability by a factor of 400 to achieve a RAW with respect to the ΔCDF of 14, placing the representative point in Region I. This corresponds to a failure rate of 0.4 per demand. As was the case in basic event (a) of Table 3, an analysis is required to determine whether this error probability is reasonable, although we expect it to be unreasonable.

Basic event (i) of Table 3, "Sump failure" has model uncertainties that have the potential to be important to the decision. There has been significant debate over even whether sufficient data exists to measure sump performance. The PRA assigns a failure probability of 5×10^{-5}. This would need to increase by a factor of 8500 to a failure probability of about 0.4 in order to impact the decision.

Looking at the basic events that have high values of RAW with respect to both CDF and ΔCDF from Table 5, we see that basic event (b) "Common-cause failure or RHR pumps" is risk-important. There are two RHR pumps. In the PRA, β is 0.15, λ_{fts} is 3.0×10^{-3}/demand, λ_{ftr} is 3.0×10^{-5}/hour and the mission time is 24 h. The common-cause failure rate is, therefore,

$$\begin{aligned}\lambda_c &= 0.15 \times [3.0 \times 10^{-3}/\text{demand} + (3.0 \\ &\quad \times 10^{-5}/\text{hour})(24 \text{ hour mission time})] \\ &= 5.58 \times 10^{-4} \text{ per mission.} \qquad (19)\end{aligned}$$

We tested the sensitivity of the CDF to variations in the common-cause failure rate and found that a factor of 20 increase would change the common-cause failure rate to 0.011 and place the representative point in Region I of the acceptance guidelines. Since λ_c is the product of two variables, we must question the values used for each variable in order to question the value used for λ_c.

We first look at the value of β. The CCFDB lists a β for RHR pump failure-while-running having a mean of 0.0464 and a 95th percentile of 0.0653. β for the failure-to-start has a mean of 0.0362 and a 95th percentile of 0.0598. The value of β used in the case study application, 0.15, is considerably higher than the CCFDB database values. Therefore, the basic event probability may be very conservative. We next look at the value of independent RHR pump failure rate. We compared the probabilities used in the case study with the probabilities used in representative PWR PRAs and found them to be consistent. Unlike the diesel generator common-cause failure basic event, where there were several model uncertainties, common-cause failure modeling is the only model uncertainty that applies to the RHR pump common-cause failure basic event. We conclude that the factor of 20 increase necessary to affect this decision is not reasonable.

There were two other basic events listed in Table 5. Basic event (a) "4160V Bus 1B fails" had no associated model uncertainties that were identified in the literature review as generically important. An analysis of basic event (c) from Table 5, "Common-cause failure of RHR heat exchangers" produced similar results to that of the RHR pump common-failure analysis and is probably not important to this licensing basis change decision.

7. Conclusions

We have sought to identify basic events where the value of their probability can change the decision, and are known to have significant model uncertainty. We focused on Level I, at power, internal events PRAs, and the decision-making process related to licensing basis changes. The acceptance guidelines with respect to a plant's CDF and ΔCDF of a proposed change have been clearly defined by RG 1.174 and the need to address all uncertainties in the decision-making process has been established. Once the basic events of interest are identified, they are analyzed to determine what their probability would need to be to affect the decision. Then, an analysis must determine whether this change is reasonable. We referred to several methods to accomplish this and provided a case study.

In our case study, a total of 12 basic events had RAW with respect to CDF showing that their uncertainty could place the licensing basis change's representative point in Region I of the acceptance guidelines, in which case the change would generally not be approved. The model uncertainties in one of these basic events have been found to be important in a review of the literature. 10 basic events have RAW with respect to ΔCDF showing that their uncertainties could place the change's representative point in Region I. Of these, three were important in the literature. Two basic events were common to both lists, showing high importance with respect to both the CDF and ΔCDF. Therefore, a total of 20 basic events were identified as important.

The decision appears to be insensitive to uncertainties in all of these basic events. In order to move the representative point into Region I, the probabilities of "failure to isolate faulty steam generator", "failure to initiate high pressure recirculation", and "failure of sump" would need to increase considerably. An evaluation of the reasonableness of these increases would be required.

We also performed a sensitivity of success criteria related to auxiliary feedwater pumps for illustration, where we changed the assumption that one feedwater was sufficient to ensure success to an assumption that either the turbine-driven pump or both motor-driven pumps were required. The alternative assumption produced a CDF of 6.85×10^{-5}, or 0.36% higher than the baseline case.

Acknowledgments

This work is a product of the project "Contributions to Risk-Informed Decision Making" and has been supported by the US Nuclear Regulatory Commission under a cooperative agreement with the MIT Department of Nuclear Science and Engineering. The views presented here are those of the authors and do not necessarily represent the views of the US Nuclear Regulatory Commission. We thank Hossein Hamzehee, Todd Hilsmeier, and Susan Cooper of the NRC Office of Nuclear Regulatory Research for their support and comments. We thank Patrick Baranowsky, Michael Cheok, Mary Drouin, Christopher Grimes, and Gareth Parry of the NRC for discussing with us issues related to risk-informed decision making. Finally, Mary Presley of MIT reviewed the manuscript and gave us useful comments.

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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE COMMISSION

In the Matter of	)	
	)	Docket No. 50-0219-LR
AMERGEN ENERGY COMPANY, LLC	)	
	)	
(License Renewal for the Oyster Creek	)	
Nuclear Generating Station)	)	June 11, 2008
	)	

CERTIFICATE OF SERVICE

I, Richard Webster, of full age, certify as follows:

I hereby certify that on June 4, 2008, I caused Citizens' Response to the Commission's order dated May 28, 2008 to be served via email and U.S. Postal Service (as indicated) on the following:

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