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November 7, 2006

VIA E-MAIL AND US-MAIL

Luis A. Reyes
Executive Director for Operations
United States Nuclear Regulatory Commission
Washington, DC 20555-0001

Dear Mr. Reyes:

I am writing on behalf of STROC, the citizen's coalition including Nuclear Information and Resource Service (NIRS), Jersey Shore Nuclear Watch, Inc., Grandmothers, Mothers and More for Energy Safety, New Jersey Sierra Club, New Jersey Environmental Federation (NJEF) and New Jersey Public Interest Research Group (NJPIRG). We are concerned that the NRC does not have reasonable assurance that the Oyster Creek Nuclear Power Plant (the "Plant") meets safety requirements. In particular, the containment system at the Plant is severely corroded and it appears that it did not even meet code requirements in 1992. According to the Atomic Energy Act, NRC is charged with protecting the health and safety of the public. 42 U.S.C. 2102. Thus, unless the NRC has reasonable assurance that the Plant meets safety requirements, it should not allow the Plant to operate.

As is clear, this letter concerns the current safety of the Plant, not relicensing. We therefore believe that it should be addressed by the Commission. However, because the Commission's letter of October 16, 2006 did not respond to Congressional enquiries regarding current safety, and your letter of October 30, 2006 did respond to those concerns, we are writing to you.

One major concern is that the severely corroded areas of the drywell shell appear to have been larger than is allowable since 1992. Recently, the reactor operator, AmerGen, stated in an e-mail to NRC Staff dated April 5, 2006 at 10 (ML060960563) that areas corroded to less than 0.736 inches in thickness "could be contiguous, *provided their total area did not exceed one square foot*" and their average thickness was greater than 0.536 inches. This statement is based on modeling conducted by General Electric ("GE") which showed that a shell with a general uniform thickness of 0.736 inches, but with a one square foot area that was 0.536 inches thick in each bay would meet code requirements. GPU Calculation No. C-1302-187-5320-024 at 9.

However, statements by NRC staff and contractors regarding the safety of the drywell

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shell at the October 3, 2006 meeting of the Advisory Committee on Reactor Safeguards License Renewal Subcommittee ("ACRS") indicated that the shell may not meet the stated requirement. Both Hans Asher and Frank Gillespie asserted that two of the 10 bays had areas of *4 square feet* that were severely corroded. Transcript of October 3, 2006 ACRS Meeting ("T.") at 153:22-24, 159:11-15, 179:9-10. Although neither staff member indicated what was meant by severe corrosion, and neither responded to enquiry made on October 5, 2006 regarding this issue, the staff's handout at the ACRS meeting indicated that the modeled thickness was 0.618 inches in Bay 13 and 0.705 inches in Bay 1. In addition, Mr. Asher stated that the inputs were based on measurements taken in 1992. T. 173:12-20. Thus, it appears that NRC Staff now believe that the thin areas of the shell in 1992 were already four times bigger than the GE modeling showed was acceptable.

When asked to respond to the concerns of six Congressmen about current safety you suggested that preliminary results from a more recent NRC study of the drywell shell indicated that the shell met code requirements. Letter from Reyes to Saxton dated October 30, 2006. If you are referring to a recent study conducted by Sandia Laboratories, this statement is highly misleading. At the ACRS meeting, one of the authors of the Sandia analysis stated that "we didn't have enough information to do the rigorous level of analysis that GE had done" and the study was *not* a look at the absolute values of the margin of safety. T. at 169:1-8. In addition, Mr. Gillespie indicated that the NRC study was "confirming their number. We're not trying to independently calculate something that's totally ours." T. 180:3-5. Moreover, the analysis presented indicated that the shell was only just above the required margin, and ACRS members indicated that they were concerned that if the location of the degraded areas moved, the safety factor would be reduced. E.g. T. 173:9-11; 163:25-164:3. Finally, even though one of the ACRS members stated "you have to be pretty thorough with your sensitivity analysis," T. 173:4-6, NRC staff failed to present any analysis of sensitivity to changes in model inputs or estimate of uncertainty of the prediction.

Thus, the NRC cannot rely on the highly uncertain preliminary results of an incomplete study that was not designed to evaluate absolute safety factors to show that the Plant meets safety requirements. Indeed, if the NRC now estimates that the areas thinner than 0.736 inches in any one bay are larger than one square foot, as staff stated to the ACRS, the NRC must not allow the Plant to restart from the current outage. Furthermore, because the Plant was at best extremely close to its required margin of safety in 1992 and we know that water has been in contact with the exterior of the shell since then, the Plant should not reopen until the measurements now being taken have been thoroughly analyzed and reviewed by NRC staff and the ACRS. Finally, even if the current results show no further deterioration since 1992 and the areas thinner than 0.736 inches are less than one square foot in each bay, further analysis is required to provide a reasonable assurance of safety because of flaws in the GE analysis pointed out by Stress Consulting, Inc. Specifically, Stress found that the GE model could have missed a critical failure mode, namely anti-symmetric buckling, and that more accurate methods are now available to calculate the current margins of safety. For ease of reference, this opinion is attached. Thus, until the structural modeling is redone in a rigorous manner taking full account of uncertainty in the inputs and is reviewed by the NRC Staff and the ACRS, the NRC cannot have any reasonable assurance that the plant meets safety requirements and should not allow the Plant to operate.

We trust you will understand that this matter is of the utmost importance for those who live close to the plant and in the region. We therefore respectfully request a written response to this letter before the plant reopens, either stating why the NRC believes that there is a reasonable assurance of safety at the Plant, or ordering AmerGen to keep the Plant off-line pending further analysis of the available data and review by the ACRS. Thank you for your consideration and we look forward to hearing from you shortly.

Yours sincerely,

Richard Webster

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c.c. Chairman Dale E. Klein
Congressman Christopher H. Smith
Congressman Jim Saxton
Congressman Robert E. Andrews
Congressman Rush Holt
Congressman Frank Pallone, Jr.
Congressman Bill Pascrell, Jr.
Governor Jon Corzine
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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.

Respectfully Submitted,

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