ArevaEPRDCPEm Resource

From: Pederson Ronda M (AREVA NP INC) [Ronda.Pederson@areva.com]
Sent: Monday, July 13, 2009 6:04 PM
To: Tesfaye, Getachew
Cc: BENNETT Kathy A (OFR) (AREVA NP INC); DELANO Karen V (AREVA NP INC); NOXON David B (AREVA NP INC)
Subject: Response to U.S. EPR Design Certification Application RAI No. 234, FSARCh. 19 (Part 1 of 4)
Attachments: RAI 234 Response US EPR DC (Part 1 of 4).pdf

Getachew,

Attached please find AREVA NP Inc.'s response to the subject request for additional information (RAI). The response document, "RAI 234 Response US EPR DC.pdf" provides technically correct and complete responses to one of the four questions. Due to E-mail size restrictions, this file is provided in 4 consecutive parts.

Appended to this file are affected pages of the U.S. EPR Final Safety Analysis Report in redline-strikeout format which support the response to RAI 234 Question 19-307.

The following table indicates the respective pages in the response document, "RAI 234 Response US EPR DC.pdf," that contain AREVA NP's response to the subject questions.

Question #	Start Page	End Page
RAI 234 — 19-304	2	2
RAI 234 — 19-305	3	3
RAI 234 — 19-306	4	4
RAI 234 — 19-307	5	51

A complete answer is not provided for three of the four questions. The schedule for a technically correct and complete response to these questions is provided below.

Question #	Response Date
RAI 234 — 19-304	October 30, 2009
RAI 234 — 19-305	October 30, 2009
RAI 234 — 19-306	October 30, 2009

Sincerely,

Ronda Pederson
ronda.pederson@areva.com
Licensing Manager, U.S. EPR Design Certification
AREVA NP Inc.
An AREVA and Siemens company
3315 Old Forest Road
Lynchburg, VA 24506-0935
Phone: 434-832-3694
Cell: 434-841-8788

From: Tesfaye, Getachew [mailto:Getachew.Tesfaye@nrc.gov]
Sent: Friday, June 12, 2009 10:23 AM
To: ZZ-DL-A-USEPR-DL
Cc: Ma, John; Xu, Jim; Clark, Theresa; Phan, Hanh; Fuller, Edward; Mrowca, Lynn; Chowdhury, Prosanta; Rycyna, John; Colaccino, Joseph; ArevaEPRDCPEm Resource
Subject: U.S. EPR Design Certification Application RAI No. 234 (2861), FSARCh. 19

Attached please find the subject requests for additional information (RAI). A draft of the RAI was provided to you on May 19, 2009, and discussed with your staff on June 11, 2009. Draft RAI Question 19-305 was modified as a result of that discussion. The schedule we have established for review of your application assumes technically correct and complete responses within 30 days of receipt of RAIs. For any RAIs that cannot be answered within 30 days, it is expected that a date for receipt of this information will be provided to the staff within the 30 day period so that the staff can assess how this information will impact the published schedule.

Thanks,
Getachew Tesfaye
Sr. Project Manager
NRO/DNRL/NARP
(301) 415-3361

Hearing Identifier: AREVA_EPR_DC_RAIs
Email Number: 651

Mail Envelope Properties (5CEC4184E98FFE49A383961FAD402D310110EF4C)

Subject: Response to U.S. EPR Design Certification Application RAI No. 234, FSARCh. 19 (Part 1 of 4)
Sent Date: 7/13/2009 6:04:06 PM
Received Date: 7/13/2009 6:04:16 PM
From: Pederson Ronda M (AREVA NP INC)

Created By: Ronda.Pederson@areva.com

Recipients:
"BENNETT Kathy A (OFR) (AREVA NP INC)" <Kathy.Bennett@areva.com>
Tracking Status: None
"DELANO Karen V (AREVA NP INC)" <Karen.Delano@areva.com>
Tracking Status: None
"NOXON David B (AREVA NP INC)" <David.Noxon@areva.com>
Tracking Status: None
"Tesfaye, Getachew" <Getachew.Tesfaye@nrc.gov>
Tracking Status: None

Post Office: AUSLYNCMX02.adom.ad.corp

Files	Size	Date & Time
MESSAGE	2643	7/13/2009 6:04:16 PM
RAI 234 Response US EPR DC (Part 1 of 4).pdf		4949253

Options
Priority: Standard
Return Notification: No
Reply Requested: No
Sensitivity: Normal
Expiration Date:
Recipients Received:

Response to

Request for Additional Information No. 234

6/12/2009

U. S. EPR Standard Design Certification
AREVA NP Inc.
Docket No. 52-020
SRP Section: 19 - Probabilistic Risk Assessment and Severe Accident Evaluation
Application Section: 19

QUESTIONS for Structural Engineering Branch 2 (ESBWR/ABWR Projects) (SEB2)

Question 19-304:

In the response to RAI 19.01-1 and 19.01-2, AREVA used the NUREG/CR-0098 median spectral shapes anchored to the average peak spectral acceleration (PSA) in 2 to 10 Hz frequency range to define the seismic margin earthquake (SME). This resulted in 1.17g average spectral acceleration for rock sites and 1.45g for the envelope of the soil sites. Comparisons of the NUREG/CR-0098 spectra with CSDRS raised by a factor of 1.76 for pga showed that the NUREG/CR-0098 spectra anchoring to the average PSA practically envelopes 1.67 times CSDRS for soil sites while for rock sites, it resulted in the spectral exceedance in the frequency range of 8 – 30 Hz. This means that the use of NUREG/CR-0098 spectral shape anchoring to the average PSA between 2 and 10 Hz results in a seismic margin less than 1.67 times CSDRS in the respective frequency range. Please also note that in the PRA-based seismic margin method, only the SME will be used for quantifying the sequence-level HCLPFs; the term: Review Level Earthquake or RLE is associated with the capacity screening of as-built and as-designed structures, systems, and components - a condition that is not available for design certifications and therefore, would not be applicable to the design certification application.

AREVA is requested to identify those SSCs on the seismic accident sequences with fundamental frequencies falling within 8 – 30 Hz, and to demonstrate that these SSCs possess adequate seismic margins to meet 1.67 times CSDRS based on the spectral acceleration. Alternatively, AREVA can use the 1.67 times CSDRS as the SME to reconstitute the sequence level HCLPFs for the US EPR design.

AREVA is requested to demonstrate the seismic margin of 1.67 time CSDRS for the Nuclear Island against the seismic induced sliding and overturning.

The staff also requests that AREVA provide a COLA Action Item for meeting Part 52.79(a)(46) to update the system model (seismic accident sequences) developed in DCD to incorporate site-specific capacity reductions due to site-specific effects (soil liquefaction, slope failure etc.) and site-specific structures (safety related site-specific intake structure, intake tunnel heat sink), if any appears on the seismic accident sequences used for the PRA-based HCLPF assessments of the DC, and demonstrate the seismic margins of the applicable site-specific SSCs; the HCLPFs for respective site-specific SSCs will be estimated based on the site-specific GMRS.

Further, Since DCD uses 1.67 x CSDRS as SME and COLA will use 1.67 x GMRS as SME for margin assessment, AREVA is requested that the HCLPFs be provided in terms of PGA for consistency.

Response to Question 19-304:

A response to this question will be provided by October 30, 2009.

Question 19-305:

Follow-up to RAI Question 19.01-8

The response to 19.01-8 is inadequate with respect to equating the ASME Service Level C/Factored load capacity to the ultimate capacity as defined in Section 3.8.1.4.11 of the U.S. EPR FSAR, rev. 0, which appears to be a fragility value. The ASME Service Level C analysis is a deterministic design process with allowable stress and strain limits specified in the ASME B&PV Code Section III. Furthermore, the response did not address the containment structural performance when subjected to internal pressurization by hydrogen released assuming 100% fuel clad-fuel reaction followed by hydrogen burning as required by Part 50.44(c)(5) for new reactor designs (a guidance is provided by RG 1.7, rev. 3). Instead, the response introduced the AICC assumption and associated pressure and temperature, but provided no discussion of how the AICC represents the accident scenario assuming 100% fuel clad-fuel reaction followed by hydrogen burning.

To facilitate the staff's review and evaluation of the U.S. EPR containment structural performance to meet Part 50.44(c)(5), AREVA is requested to provide the following information:

1. An analysis which estimates the containment internal pressure load time history due to the hydrogen released by assuming 100% fuel clad-fuel reaction followed by hydrogen burning;
2. A structural analysis of the containment subject to the pressure load as determined in step 1 plus the dead load (a guidance is provided in RG 1.7, rev. 3); the Code specified minimum material properties at the temperature should be used in the analysis (however, the temperature load should not be included).
3. Demonstrate that the containment response in terms of the liner strain determined from the step 2 analysis remains below the ASME Service Level C limit.

Response to Question 19-305:

A response to this question will be provided by October 30, 2009.

Question 19-306:

Follow-up to RAI Question 19.01-9

To address the SECY-93-087 containment deterministic structural performance expectation, the applicant stated in the RAI response that “Relevant scenarios in the U.S. EPR are defined as those having a Core Damage Frequency (CDF) greater than 1.0E-8/yr. In AREVA’s scenario identification analysis, this 1.0E-8/yr threshold captured categories of events covering over 95% of the CDF.” Since accident sequences which comprise 95% of the CDF should encompass the events most likely to challenge the containment structural integrity, the staff accepts the more likely accident scenarios defined in this manner for the containment deterministic structural performance evaluation. However, the applicant did not provide the controlling containment pressure and temperature load time histories for the identified accident events, nor was the containment Service Level C capacity correctly addressed for reasons stated in the 19.01-8 supplement.

To facilitate the staff’s review and evaluation of the U.S. EPR containment deterministic structural performance evaluation, AREVA is requested to provide the following information:

1) Provide the controlling containment pressure demand in terms of pressure time history and the corresponding temperature time history derived from the more likely accident scenarios.
2) A structural analysis of the containment subject to the pressure load as determined in step 1 plus the dead load; the Code specified minimum material properties at the temperature should be used in the analysis (however, the temperature load should not be included in the containment structural analysis).
3) Demonstrate that for the initial 24 hours following the onset of core damage, the containment response in terms of the stresses or strains for containment structural elements as determined from the step 2 analysis remains below the ASME Service Level C (or Factored Load) limit.
4) For the period following 24 hours after the core damage, either demonstrate that the pressure and temperature time histories are not greater than those during the initial 24-hour period, or perform additional nonlinear containment structural analysis to demonstrate that the containment still provides a barrier against the uncontrolled release of fission products.

Response to Question 19-306:

A response to this question will be provided by October 30, 2009.

Question 19-307:

Follow-up to RAI Question 19.01-10

In the response to RAI 19.01-10, the applicant stated that the composite fragility was calculated based on six containment failure modes. Based on the equation provided for determining failure probability, any additional failure modes identified for the containment would increase the containment failure probability. The applicant did not provide any justification for excluding other possible failure modes.

The applicant set the failure criterion for the tendons at 3% strain; however, the applicant did not explain what it means in the context of probability for developing the containment fragility nor did it define what the failure means for other structural components. The manner in which the information is presented in Table RAI 19.1-10-1 and Table RAI 19.1-10-2 was very confusing. Aside from that, the modeling uncertainty was not even mentioned.

To facilitate the staff's review and evaluation of the U.S. EPR containment pressure fragility evaluation, AREVA is requested to provide the following information:

a) The containment pressure fragility should be determined based on analyses which utilize appropriate material constitutive relations, and an assessment of uncertainties within a probabilistic framework. The uncertainties in the analysis results should be associated with the modeling and analysis approach (epistemic uncertainty), the material properties (aleatoric uncertainty) of the structure at the time of the accident, failure criteria or limit states used in establishing the pressure capacity, and the loading conditions that lead to pressurization of the containment.

b) Failure criteria are defined to establish limit states on the structural response where the internal pressure is no longer contained by the structure. Uncertainty in defining these failure criteria should be addressed; one could use median and 95% confidence values to evaluate the effect of the uncertainty on the analysis results.

c) Accident conditions leading to over-pressurization will also include elevated temperatures. Because of thermal induced stresses and material property degradation at elevated temperatures, the fragility for over-pressurization is also a function of temperature. Thus, the fragility analyses should be conducted for three different thermal conditions, 1) steady state normal operating temperatures (referred to as ambient conditions), 2) steady state conditions representing long-term accident conditions, and 3) transient thermal conditions such as a temperature spike representative of direct containment heating conditions.

Uncertainty associated with the modeling used in the analyses for determining the failure pressures should be addressed. This uncertainty concerns the finite element mesh discretization, the type of element formulations used, the robustness of the constitutive models, the equilibrium iteration algorithms and convergence tolerances, geometric imperfections, fabrication and construction exactness, rebar placement locations, and the like. This modeling uncertainty should be quantified as part of the fragility calculation.

Response to Question 19-307:

The results presented in this response are based on a revised analysis of the ultimate strength capacity of the U.S.EPR reactor containment. As described in the Response to RAI 155, Question 03.08.01-10, this revision updated the methodology for determining the containment ultimate capacity margins to meet the requirements of SRP 3.8.1 (Reference 1).

The U.S. EPR containment is sub-divided into six areas, which, when assembled together represent a continuous model of the entire containment structure. These areas are the cylindrical wall, spherical dome, dome belt, gusset (the connection from the wall to the foundation) and the equipment hatch (horizontal and vertical sections). Major penetrations, and personnel and emergency airlocks are not currently modeled because design details for these will be developed later in the design process. The boundaries of the six areas modeled are selected so that each area covers a pressure boundary location in the containment, and for each one the most limiting failure mode is analyzed. Table19-307-1 summarizes the controlling failure mode for each area of the containment. Figure 19-307-1 shows the entire containment model developed with ANSYS (Reference 2).

The equation presented in the Response to Question 19.01-10 for calculating the composite fragility curve of the containment represents the sum of the probabilities of failure of the six areas previously described, initially called “locations”. In fact, the equation corresponds to one of the two methods used in the Level 2 probabilistic risk assessment (PRA) containment fragility analysis to represent the composite fragility curve. The second method uses a Monte Carlo sampling to select the highest probability of failure from the six areas as representative of the composite fragility curve for a given pressure. The two methods were found to be equivalent and were used interchangeably under the conditions of the present analysis. However, the second method is preferred because it is not impacted by the failure dependencies between the six different areas considered; rather it translates the sensitivity of the containment fragility to the weakest locations.

The failure areas not included in this analysis (i.e., personnel and emergency airlocks and containment penetrations) are local disturbances of the containment that have been extensively analyzed in the standard EPR design. It was concluded that these structures should only be considered for containment leakage analyses, and that the anticipated leakage would not preclude containment rupture. Therefore, these additional locations can be excluded from the evaluation of the containment failure due to rupture.

The methodology used in the U.S. EPR containment fragility analysis as well as the results, are further detailed below.

The structural analysis of the containment fragility evaluates the median and ultimate pressure capacities for the six containment areas identified above. For each containment area, the ultimate pressure capacity is evaluated through a finite-element analysis within the context of a deterministic approach. While the median pressure is calculated using an alternate methodology based on correlations that account for the strength of the material and the proportion to which they contribute to the composition of a given area. This analysis is performed for a design basis loss of coolant accident (LOCA) at a steady state temperature of 309°F (154°C) inside containment that is bounding to the ambient temperature case, taking into account a thermal gradient through the containment wall. The two methodologies are described as follows:

- The ultimate pressure capacity of the U.S. EPR containment is performed using a deterministic approach in conformance with Standard Review Plan SRP 3.8.1. A nonlinear finite element analysis is performed for the pre-stressed containment components including building structure and steel liner. Tendon and passive reinforcement are modeled as a membrane corresponding to the reinforcement location using ANSYS SHELL43 elements (four nodes of plastic, large strain shell elements). An ANSYS SHELL43 element is also used on the liner plate using isotropic material properties. Geometric and material nonlinearity (elastic-perfectly plastic material) are accounted for with the large displacement option turned on in ANSYS. The finite element mesh is shown in Figure 19-307-1 (the middle portion of the cylinder wall is not shown for clarity purposes). Nonlinear finite element analysis is performed in five load step runs; load steps 1 to 3 apply initial loads (i.e., dead weight and prestress) and load steps four to five apply the accident loads (accident temperature followed by incremental internal pressure up to the ultimate capacity).
- The median pressures of failure, as calculated for two of the six areas (the cylinder wall and spherical dome), are determined through the evaluation of the material strength against a pre-defined failure criterion that will be described below. The median pressure capacity Pm together with the total uncertainty β is used to derive P95 the “95% non-exceedance pressure”. These three parameters represent respectively the median, the lognormal standard deviation and the fifth percentile of the lognormal distribution representing the fragility of a given area of the containment.

The ultimate pressure capacity is lower than the median pressure capacity and in some cases was used as a substitute to the 95 percent non-exceedance pressure. This was the case for two of the six areas considered in this analysis (dome belt and gusset). Then, using the 95 percent non-exceedance pressure and the total uncertainty β the median pressure capacity is derived. These values were derived using the following relations:

$$P_{95} = P_m \exp(-1.65\beta) \text{ (Eq1) and } \beta^2 = \beta_S^2 + \beta_M^2 \text{ (Eq2)}$$

where: β_S is the material uncertainty also called variability

β_M is the modeling uncertainty

The combined uncertainty of either β_S or β_M due to the contribution of different material components in any section was obtained using the following equation:

$$\beta = \sqrt{\sum_i (\alpha_i \beta_i)^2} \text{ (Eq3)}$$

where: α_i is the contribution in percentage of “i” material component

β_i is the uncertainty of “i” material component.

The following are the steps of the methodology used in the U.S. EPR structural capacity evaluation:

- Identification of the list of containment sub-areas to be evaluated (discussed above).
- Definition of the limiting criteria for ultimate capacity based on recommendations of SRP 3.8.1 and NUREG/CR-6096.
- Calculations of material strength and total uncertainty.
- Finite element analyses for all identified sub-areas.
- Determination of ultimate pressure capacity (Pu) and the associated margin of safety for design basis accident temperature for each sub-component

Failure Criteria

The failure criterion used for the standard EPR design containment analysis, defined as; "the ultimate strength for the post-tension strands," is estimated at an average breaking strain of three percent. In the U.S. EPR analysis, a membrane hoop strain limit of 0.8 percent for estimating capacity of pre-stressed concrete containments was used in the finite element analysis as recommended in SRP 3.8.1. This strain limit is applicable away from structural discontinuities and applies to all materials resisting internal pressure (for example, tendons, rebar and the liner plate). For consistency, the strain limit of one percent was adopted for the alternate methodology that is based on strength correlations.

Material strength

The statistical parameters used for the standard EPR containment fragility evaluation are based on generic data from past studies on European Nuclear Power Plants as well as specific EPR data.

The material properties used for the U.S. EPR Reactor Containment Building (RCB), the yielding stress and strain of the liner, passive reinforcement and tendon, and the tendon and concrete ultimate stress and strain, are very similar to those used in the standard EPR design. Therefore, similar material strength uncertainties, βs, established for the standard EPR were used for the U.S. EPR, as summarized in Table 19-307-2 for an ambient temperature of 70ºF.

The material properties under steady state thermal conditions corresponding to LOCA scenarios with an inner surface temperature of 309ºF are summarized in Table 19-307-3. Supporting studies have shown that the wall temperature drops to 176ºF (80ºC) at the inner rebar location and it further drops to 86ºF (30ºC) at the exterior concrete surface of the cylindrical containment structure. The temperature at the exterior concrete surface of the dome drops to 102ºF (39ºC). NUREG/CR-6906 observes that temperature up to 400°F has only a small effect on the ultimate pressure capacity of the containment since the cracked concrete carries no tension regardless of temperature. It also states that temperature up to 400°F has only a minor effect on typical rebar properties. Therefore, due to LOCA, 309°F is considered at the liner, concrete temperature is conservatively assumed to be 176°F and the rebar and tendon material properties are not affected by the elevated temperature during a LOCA.

The modeling uncertainties in the U.S. EPR capacity evaluations are assumed to be the same as those used for the standard EPR analysis, based on the similarity of the overall configuration of the RCB. The values used are summarized in Table 19-307-4.

The detailed evaluation of the median pressure capacity for each one of the six areas described above is provided in the following discussion. As previously mentioned, the median pressures of failure for the cylinder wall and the spherical dome were calculated from the strength correlations The median pressure of failure for the dome belt and gusset was obtained by taking the ultimate pressure (derived from the finite element study) as the 95 percent non-exceedance pressure, and by deriving the median pressure using equation1.

The correlations used to derive the median pressure (Pm) based on the median material strengths and compositions used only membrane failure for the cylinder wall and the spherical dome. These correlations are:

For the cylinder wall $P_m = \frac{Nult}{Ri}$ (Eq4)

For the dome $P_m = 2\left(\frac{Nult}{Ri}\right)$ (Eq5) and $Nult = \sum_i S_i * \sigma_i$ (Eq6)

where: $Nult$ is the ultimate force of constituent i

S_i is the cross section area for constituent i

σ_i is the median stress for constituent i

R_i is the radius of the inner containment face

Detailed Analysis of the Six Containment Areas:

- Cylinder wall:

Table 19-307-5 summarizes the cylinder wall parameters used to evaluate the “hoop membrane” failure mode for the cylinder wall area.

With a total ultimate force, N_{ult} of 3283 kip/ft and a radius R_i, of the inner face of containment of 76.77 feet, the resulting median pressure is $P_m = N_{ult}/R_i$ = 3283/76.77 ksf = 42.76 ksf = 297 psig

Materials Uncertainty $ß_s$

The combined uncertainty from the four constituents is obtained using equation 3, the participation to the total ultimate force is obtained from table 19-307-5, and the constituent uncertainty is obtained from Table 19-307-3.

$$\beta_s = \sqrt{\sum_i (\alpha_i \beta_{si})^2} = \sqrt{(0.03 \times 0.12)^2 + (0.27 \times 0.05)^2 + (0.70 \times 0.02)^2} = 0.02$$

Modeling Uncertainty, $ß_m$

The same equation and participation to the total ultimate force fractions as for the material uncertainty are used. The modeling uncertainties for each constituent are obtained from Table 19-307-4.

$$\beta_m = \sqrt{\sum_i (\alpha_i \beta_{mi})^2} = \sqrt{(0.03 \times 0.051)^2 + (0.27 \times 0.047)^2 + (0.70 \times 0.018)^2} = 0.018$$

The overall uncertainty ß is:

$$\beta_s = \sqrt{\beta_s^2 + \beta_m^2} = \sqrt{0.02^2 + 0.018^2} = 0.027$$

The “95% non-exceedance pressure”, P95, of the cylinder wall is found to be:

$$P_{95} = P_m e^{(-1.65\beta)} = (297)e^{(-1.65 \times 0.027)} = 284 \text{ psig}$$

The ratio of the 95 percent non-exceedance pressure to the design pressure is:

$$P_{95}/P_{des} = 284/62 = 4.58$$

From the finite element analysis results the ultimate pressure capacity for a typical zone of the dome is Pu = 267 psig, which is based on the deterministic approach estimating the ultimate capacity of a pre-stressed concrete containment. Since the median material strength is higher than the guaranteed material strength used in the deterministic approach, it is expected that the median pressure capacity (Pm = 297 psig) calculated using the median stresses is found to be higher than that (Pu = 267 psig) determined by the finite element analysis. The results of finite element analysis for the cylinder wall are shown in Figures 19-307-2 through 19-307-10. These plots show an ultimate pressure capacity of 267 psig at time step 271, corresponding to a strain criteria of approximately 0.8 percent..

- Spherical dome:

Using the same methodology described above, the “membrane” failure mode of the spherical dome area is investigated. The median capacity is determined based on the parameters summarized in Table 19-307-6.

The total ultimate force, N_{ult} is 2104 kip/ft. With a radius, R_i, of the inner face of the containment of 105 feet the median pressure is $P_m = 2(N_{ult}/R_i) = 2\times2104/105 = 40.08$ ksf = 278 psig.

<u>Materials Uncertainty, $ß_s$</u>

The combined uncertainty of the constituents is obtained using equation 3, the participation to the total ultimate force is obtained from Table 19-307-6 and the constituent uncertainty is obtained from Table 19-307-3.

$$\beta_s = \sqrt{\sum_i (\alpha_i \beta_{si})^2} = \sqrt{(0.05 \times 0.12)^2 + (0.21 \times 0.05)^2 + (0.74 \times 0.02)^2} = 0.019$$

<u>Modeling Uncertainty, $ß_m$</u>

The modeling uncertainties for the membrane failure are the same as the values used for the standard EPR. The combined uncertainty is also obtained using equation 3, the participation to the total ultimate force is obtained from Table 19-307-6, and the modeling uncertainties are obtained from Table 19-307-4.

$$\beta_m = \sqrt{\sum_i (\alpha_i \beta_{mi})^2} = \sqrt{(0.05 \times 0.051)^2 + (0.21 \times 0.047)^2 + (0.74 \times 0.018)^2} = 0.017$$

The overall uncertainty ß is calculated by the equation:

$$\beta_s = \sqrt{\beta_s^2 + \beta_m^2} = \sqrt{0.019^2 + 0.017^2} = 0.026$$

The “95% non-exceedance pressure”, P95, for the dome is found to be:

$$P_{95} = P_m e^{(-1.65\beta)} = (278)e^{(-1.65 \times 0.026)} = 266 \text{ psig}$$

The ratio of the 95 percent non-exceedance pressure to the design pressure is:

P_{95}/P_{des} = 266/62 = 4.29

From the finite element analysis results the ultimate pressure capacity for a typical zone of the dome is Pu = 249 psig, as above it was also expected that the median pressure capacity (Pm = 278 psig) calculated using the median stresses be higher than that (Pu = 249 psig) determined by the finite element analysis. The results of the finite element analysis of the dome belt are shown in Figures 19-307-11 through Figures 19-307-19. These plots show an ultimate pressure capacity of 249 psig at time step 253, corresponding to a strain criteria of approximately 0.8 percent.

- Dome belt:

The 95 percent non-exceedance pressure (P95), obtained from the finite element analysis is 173 psig. After the uncertainty was evaluated, the median was derived from the statistical relations between Pm, P95 and the uncertainty.

The results of the finite element analysis of the dome belt are shown in Figures 19-307-20 through Figures 19-307-26. These plots show an ultimate pressure capacity of 173 psig at time step 177, corresponding to strain criteria of approximately 0.8 percent.

Material and modeling uncertainties:

As shown in Table 19-307-5 and Table 19-307-6 for the cylinder wall and dome respectively, the contribution of the liner plate to the containment strength is less than 5 percent, therefore the contribution of liner plate to βs and βm can be ignored.

Since a reverse process is adopted to find median pressure capacity the minimum βs and βm values of tendon and reinforcement are conservatively assumed to be βs = 0.02, and βm = 0.02. The overall uncertainty ß is calculated by the equation:

$$\beta_s = \sqrt{\beta_s^2 + \beta_m^2} = \sqrt{0.02^2 + 0.02^2} = 0.028$$

The median pressure capacity, Pm, is found to be:

$$P_m = P_{95} / e^{-1.65\beta} = 173 / e^{(-1.65\times0.028)} = 181 \text{psig}$$

- Gusset:

The same methodology as for the dome belt, using the finite element deterministic methods to find the “95% non-exceedance” pressure capacity first then the median pressure is applied. P95 is found to be 315 psig from the finite element.

The results of the analysis of the gusset (base of the cylinder wall) are shown in Figures 19-307-27 through 19-307-33. These plots show an ultimate pressure capacity of 315 psig at time step 319, corresponding to strain criteria of approximately 0.8 percent.

Material and modeling uncertainties:

The same methodology and values as for the cylinder wall were used, with a total uncertainty of 0.028. The median pressure capacity, Pm, is found to be:

$$P_m = P_{95} / e^{-1.65\beta} = 315 / e^{(-1.65\times0.028)} = 330 \text{psig}$$

- Flexural failure around the equipment hatch:
  - Hatch V2 (vertical section).

The median pressure for the standard EPR design Pm is 229 psig. With much more reinforcement placed for the U.S. EPR, the median pressure capacity for U.S. EPR is determined to be equal to or greater than Pm = 229 psig.

  - Hatch H2 (horizontal section).

The median pressure for the standard EPR design Pm is 296 psig. Similarly to the vertical section of the hatch, the additional reinforcement of the U.S.EPR results in a U.S. EPR Pm of at least 296 psig.

Material and modeling uncertainties:

As discussed above, based on Tables 19-307-5 and 19-307-6, the contribution of concrete and liner plate to ßs and ßm can be ignored. A conservative value of material uncertainty $ß_s$ of 0.05 is assumed. A modeling uncertainty value of 0.05 is also assumed for $ß_m$ for the tendon and reinforcement including the sections of the resistive elements. Finally, a conservative value of 0.05 is assumed for the other modeling uncertainties $ß_{m2}$.

$ß_s$ = 0.05 $ß_m$ = 0.05 and $ß_{m2}$ = 0.05

Therefore, overall uncertainty is equal to

$$\beta = \sqrt{\beta_s^2 + \beta_m^2 + \beta_{m2}^2} = \sqrt{0.05^2 + 0.05^2 + 0.05^2} = 0.09$$

This uncertainty applies to both the vertical section V2 and the horizontal section H2.

Summary

Table 19-307-7 presents a summary of the median pressure capacities of the six areas that represent the entire containment model with the associated 95 percent non-exceedance pressures and the total uncertainties, the ratio of the 95 percent non-exceedance pressure (P95) to the design pressure Pdes is also provided.

The values summarized in Table 19-307-7 are used in the Level 2 PRA to generate the revised fragility curves of each area analyzed as well as the composite fragility curve for the containment at 309 ºF steady state thermal condition. The containment composite fragility curve generated through the use of a Monte Carlo sampling combines the results from each of the six individual failure areas into a single distribution representing the capacity. The fragility curves for the individual six failure areas and the composite are presented respectively in Figures 19-307-34 and 19-307-35. It should be noted that the containment failure is driven by the dome belt failure, and as such the composite fragility curve is equivalent to the dome belt fragility curve. The change in the dome belt failure pressure resulting from the revision of the structural analysis of the U.S.EPR containment fragility is less than 4 percent. This change does not impact the results of the Level 2 phenomenological evaluation as the pressure loads considered are typically lower than the median failure pressure. The fragility curves resulting from the present analysis have been integrated into the Level 2 PRA and U.S. EPR FSAR Tier 2, Section 19.1.4.2.1.3, Table 19.1-21 and Figure 19.1-8 are revised accordingly.

References for Question 19-307:

1. NUREG-0800, "Standard Review Plan 3.8.1, Concrete Containment", Revision 2, U.S. Nuclear Regulatory Commission, March 2007
2. ANSYS V10.0 SP1.

FSAR Impact:

U.S. EPR FSAR Tier 2, Section 19.1.4.2.1.3, Table 19.1-21 and Figure 19.1-8 will be revised as described in the response and indicated on the enclosed markup.

Table 19-307-1—U.S. EPR Containment Sub-areas and the Associated Failure Modes

Containment Area	Failure Mode
Cylinder wall	Hoop membrane failure
Spherical dome	Membrane failure
Dome belt	Flexural failure
Gusset (Base of cylinder wall)	Flexural failure
Equipment hatch[1] (horizontal section)	Flexural failure
Equipment hatch (vertical section)	Flexural failure

Note:

1. It is noted that the evaluation does not consider the steel equipment hatch cover or cylinder.

Table 19-307-2—U.S.EPR Containment Material Parameters at 70ºF

U.S. EPR	Yield Strength			Ultimate strength		
Constituent	Guaranteed Stress ksi	Strain Limit %	Uncertainty βs	Guaranteed Stress ksi	Strain Limit %	Uncertainty βs
Liner	38	0.20	0.12[1]	70	17	-
Passive Reinforcement	60	0.20	0.05	90	7	-
Tendon	243 (90% of ultimate stress)	1.0	0.02	270	3.5	0.02
Concrete	-	-	-	7.0[2]	0.3[3]	0.14[4]

Notes:

1. For liner plate material, βs = 0.12 for A-36 material is adopted from EPRI TR-103959.
2. Concrete compressive strength, f'c, for the containment base is 4000 psi.
3. The maximum strain at extreme concrete compression fiber shall be assumed equal to 0.003.
4. A minimum value of 0.14 is adopted, which coincides with what was used for the standard EPR design.

Table 19-307-3—U.S.EPR Containment Material Properties at Elevated Temperatures Representing a LOCA Scenario

Constituent	Uncertainty βs	Guaranteed Stress (ksi)	Median Stress (ksi)
Liner Plate at 309ºF	0.12	33.5	40.8
Passive reinforcement at 70ºF	0.05	60	65.2
Tendons at 70ºF	0.02	243	251.2
Concrete at 176ºF	0.14	6.86	8.64

Table 19-307-4—U.S.EPR Containment Constituents Modeling Uncertainties

Constituent	Modeling Uncertainty $ß_m$
Liner Plate	0.051
Rebar	0.047
Tendons	0.018

Table 19-307-5—Ultimate Hoop Membrane Forces in Cylindrical Wall

Constituents	Cross Section Area (in^2/ft)	Median Stress (ksi)	Ultimate Force (kip/ft)	Participation to Total Ultimate Force (αi)
Liner Plate	2.63	40.8	107	3%
Reinforcing Bars	13.5	65.2	880	27%
Tendons	9.14	251.2	2296	70%
Total			3283	100%

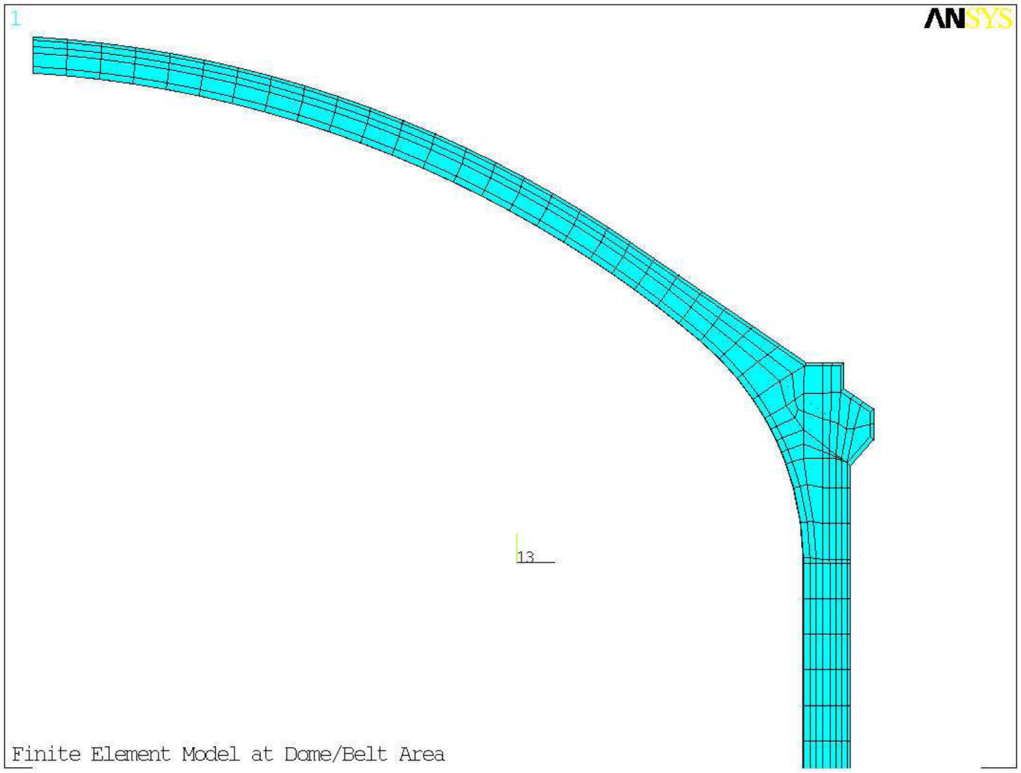
Table 19-307-6—Ultimate Membrane Forces in Dome Typical Section

Consituents	Cross Section Area (in^2/ft)	Median Stress (ksi)	Ultimate Force (kip/ft)	Participation to Total Ultimate Force (αi)
Liner Plate	2.63	40.8	107	5%
Reinforcing Bars	6.75	65.2	440	21%
Tendons	6.2	251.2	1557	74%
Total			2104	100%

Table 19-307-7—Containment Pressure Capacity under a LOCA temperature condition of 309ºF

Failure area	Total uncertainty ß	Pm Median Pressure (psig)	P95 95% Non-exceedance Pressure (psig)	Ratio P95/Pdes
Cylinder wall	0.027	297	284	4.58
Spherical dome	0.026	278	266	4.29
Dome belt	0.028	181	173	2.79
Gusset (Base of cylinder wall)	0.028	330	315	5.08
Equipment hatch (vertical section V2)	0.09	229	197	3.18
Equipment hatch (horizontal section H2)	0.09	296	255	4.12

Figure 19-307-1—U.S. EPR Containment Finite Element Model



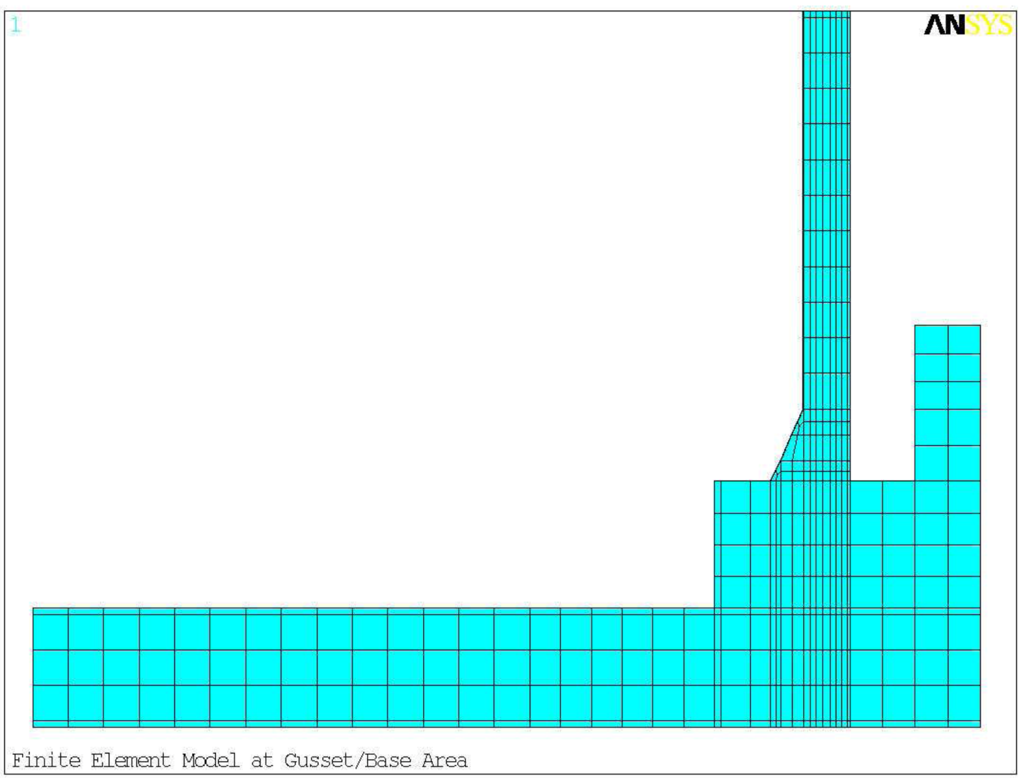


Figure 19-307-2—Concrete Hoop Stress (S_y) in Cylinder

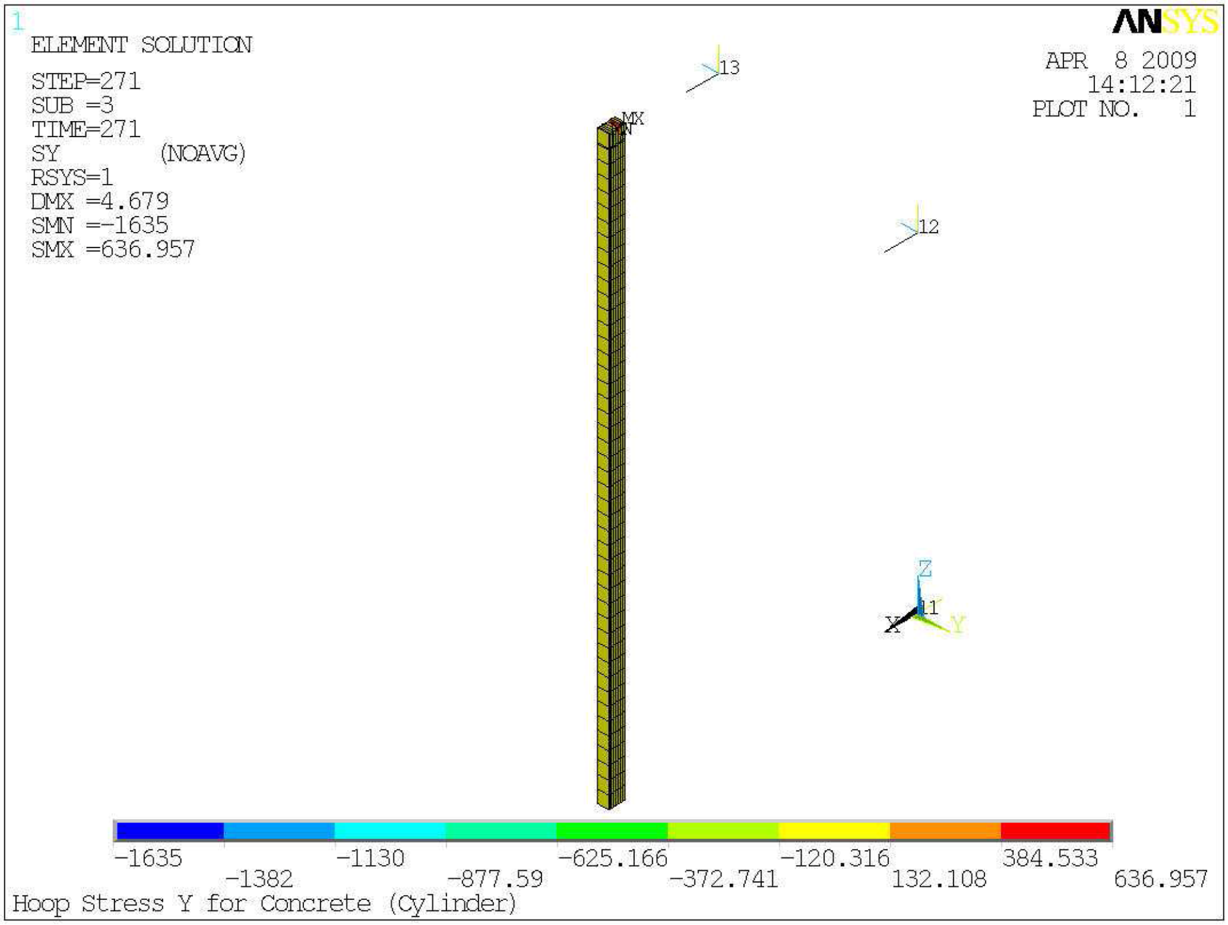


Figure 19-307-3—Concrete Meridional Stress (S_z) in Cylinder

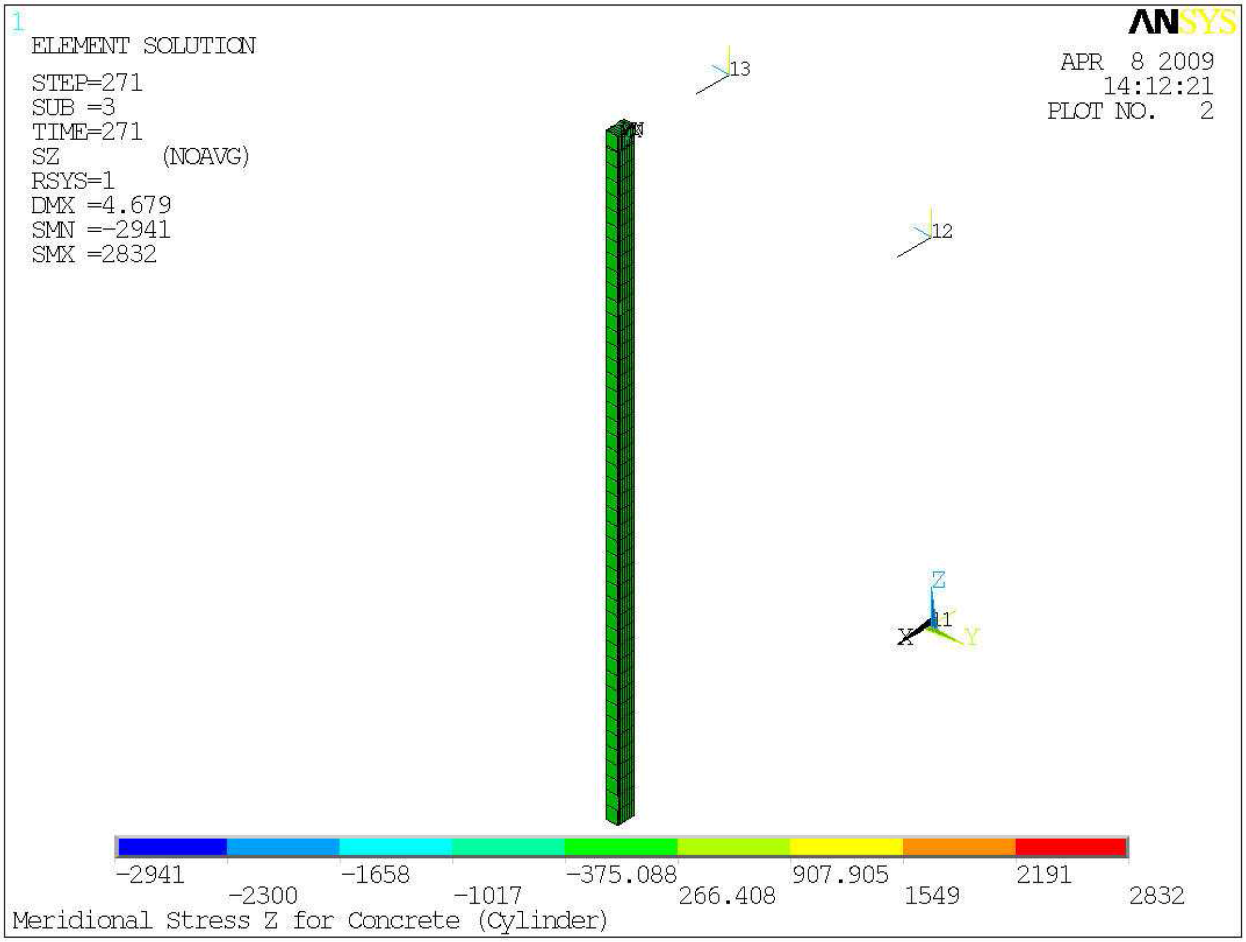


Figure 19-307-4—Concrete Principal Stress (S_3) in Cylinder

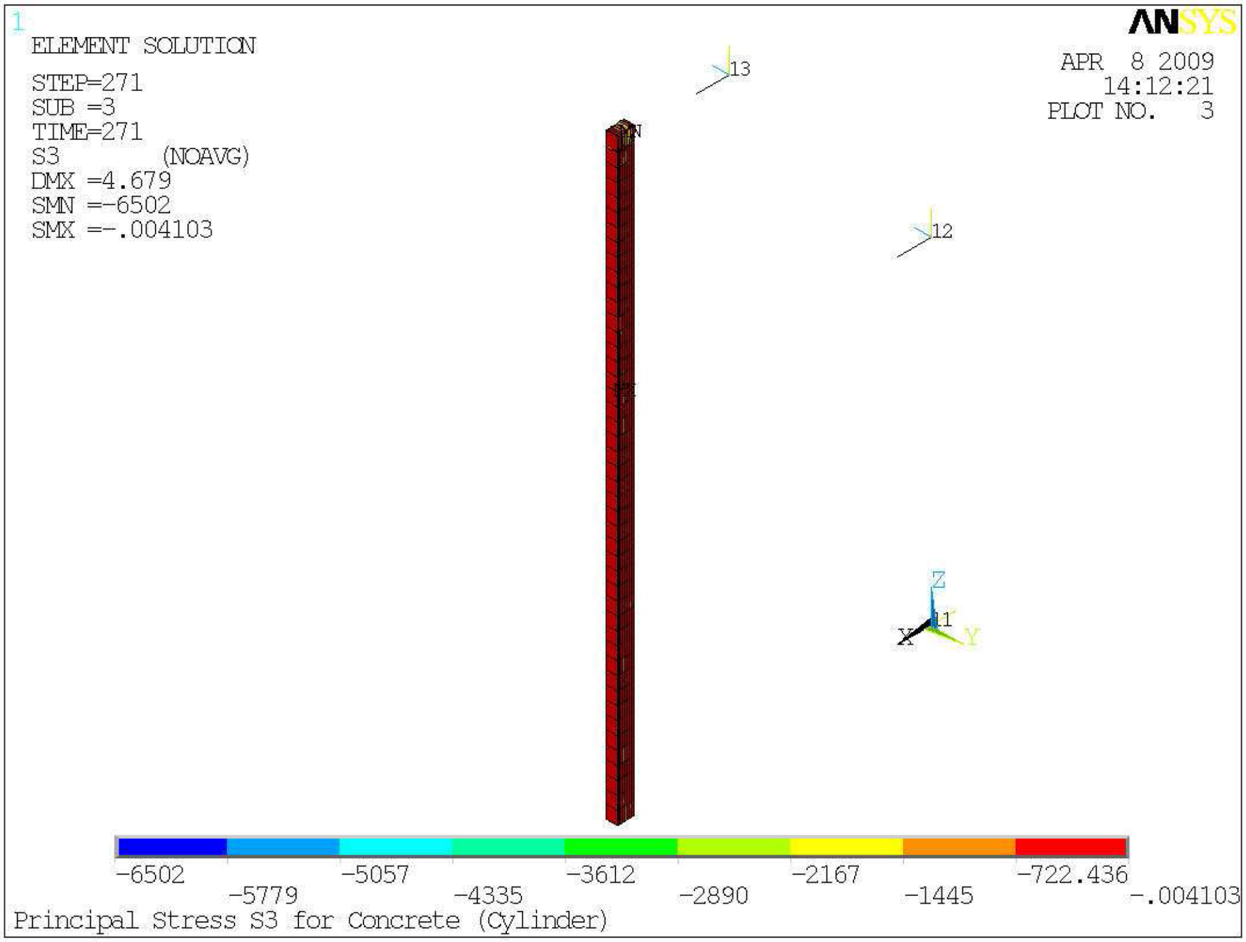


Figure 19-307-5—Rebar Hoop Strain (ε_y) in Cylinder

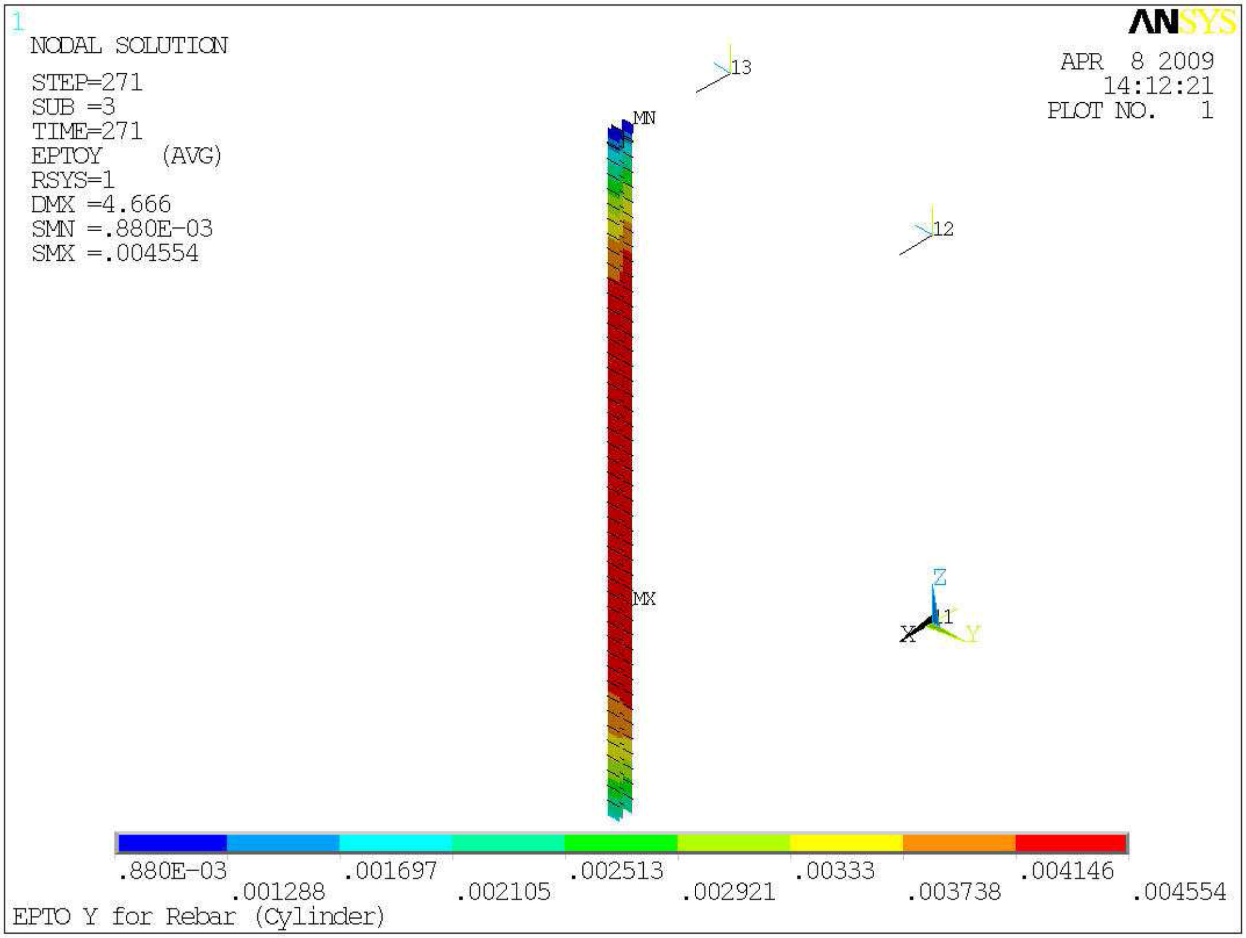


Figure 19-307-6—Rebar Meridional Strain (ε_z) in Cylinder

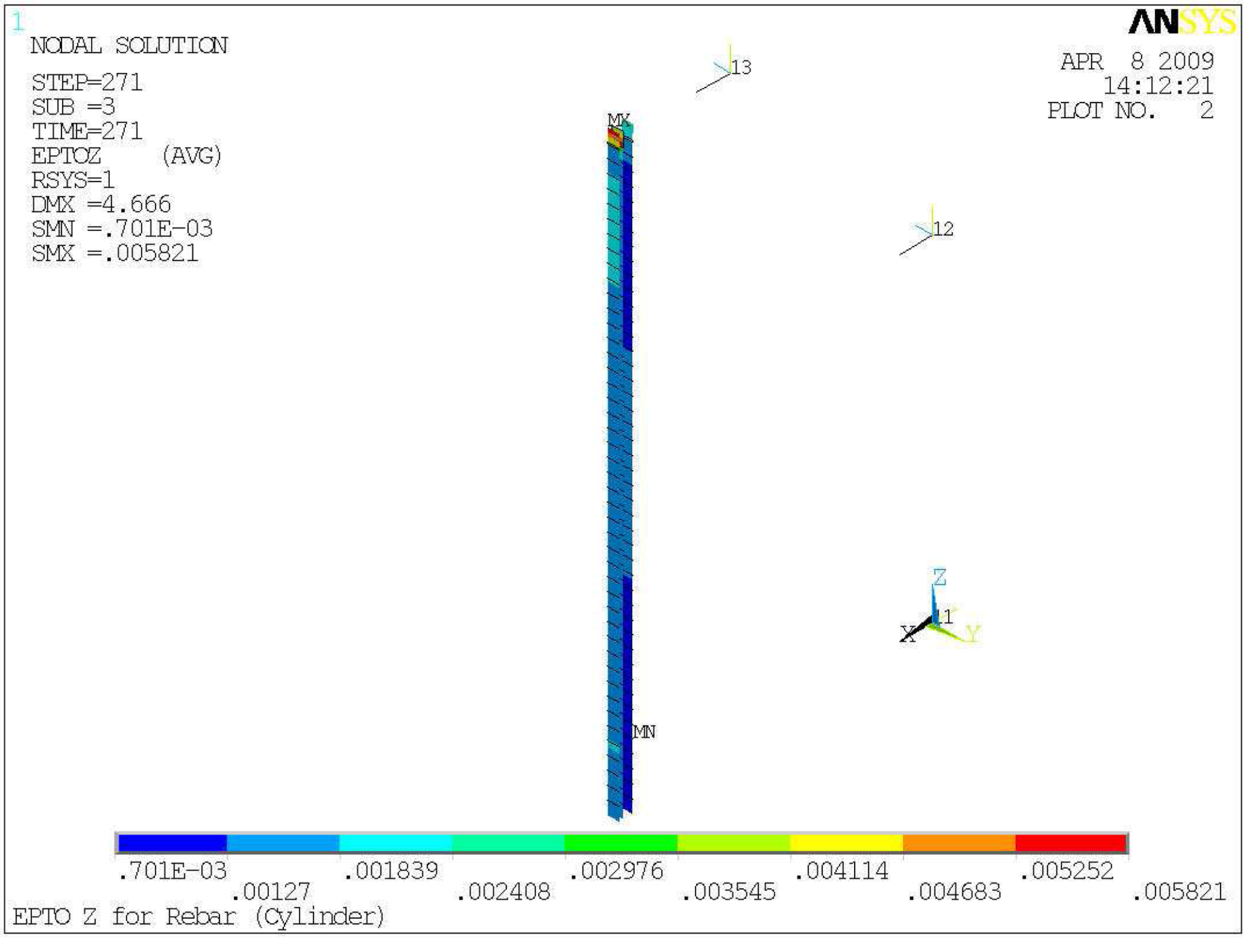


Figure 19-307-7—Liner Plate Hoop Strain (ε_y) in Cylinder

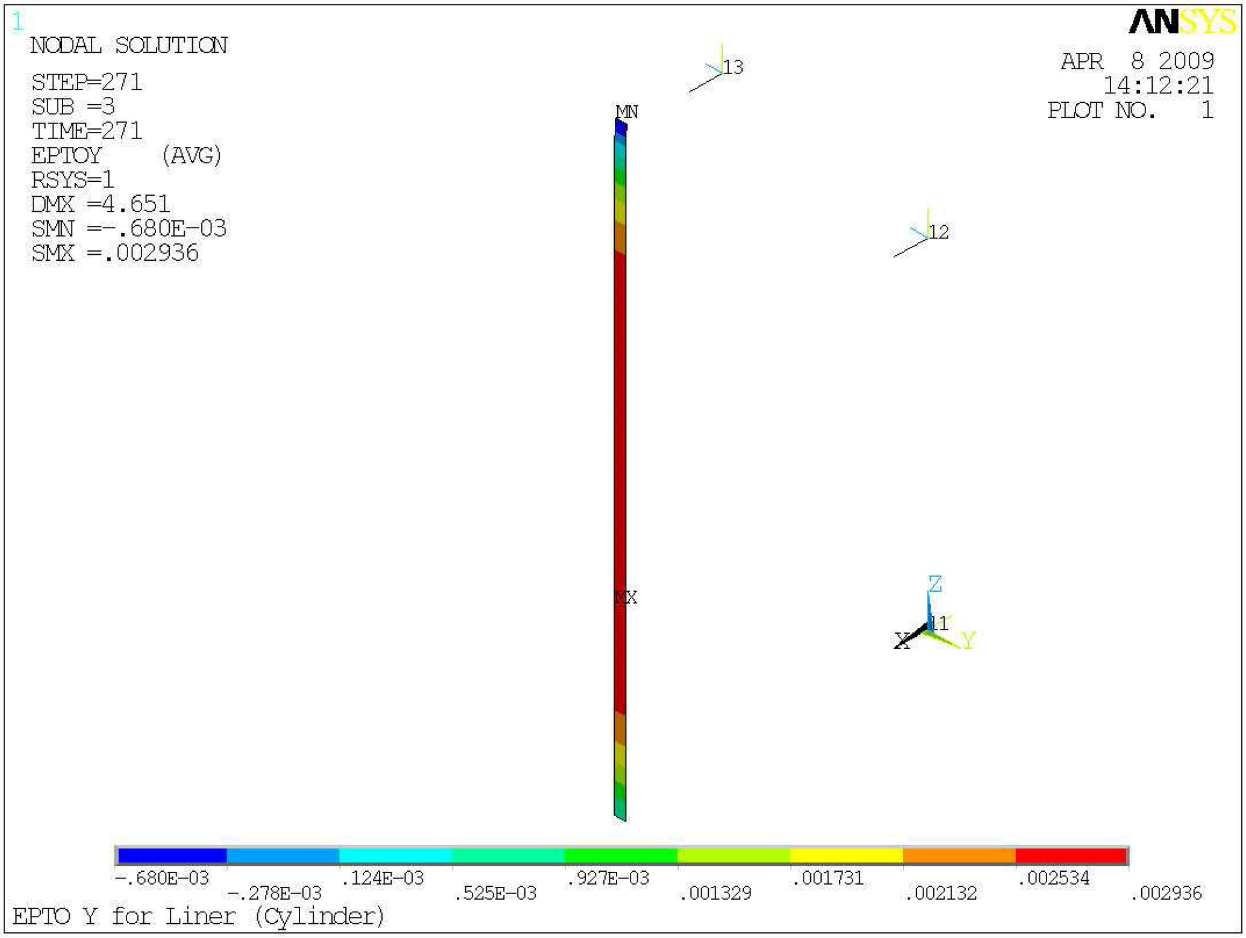


Figure 19-307-8—Liner Plate Meridional Strain (ε_z) in Cylinder

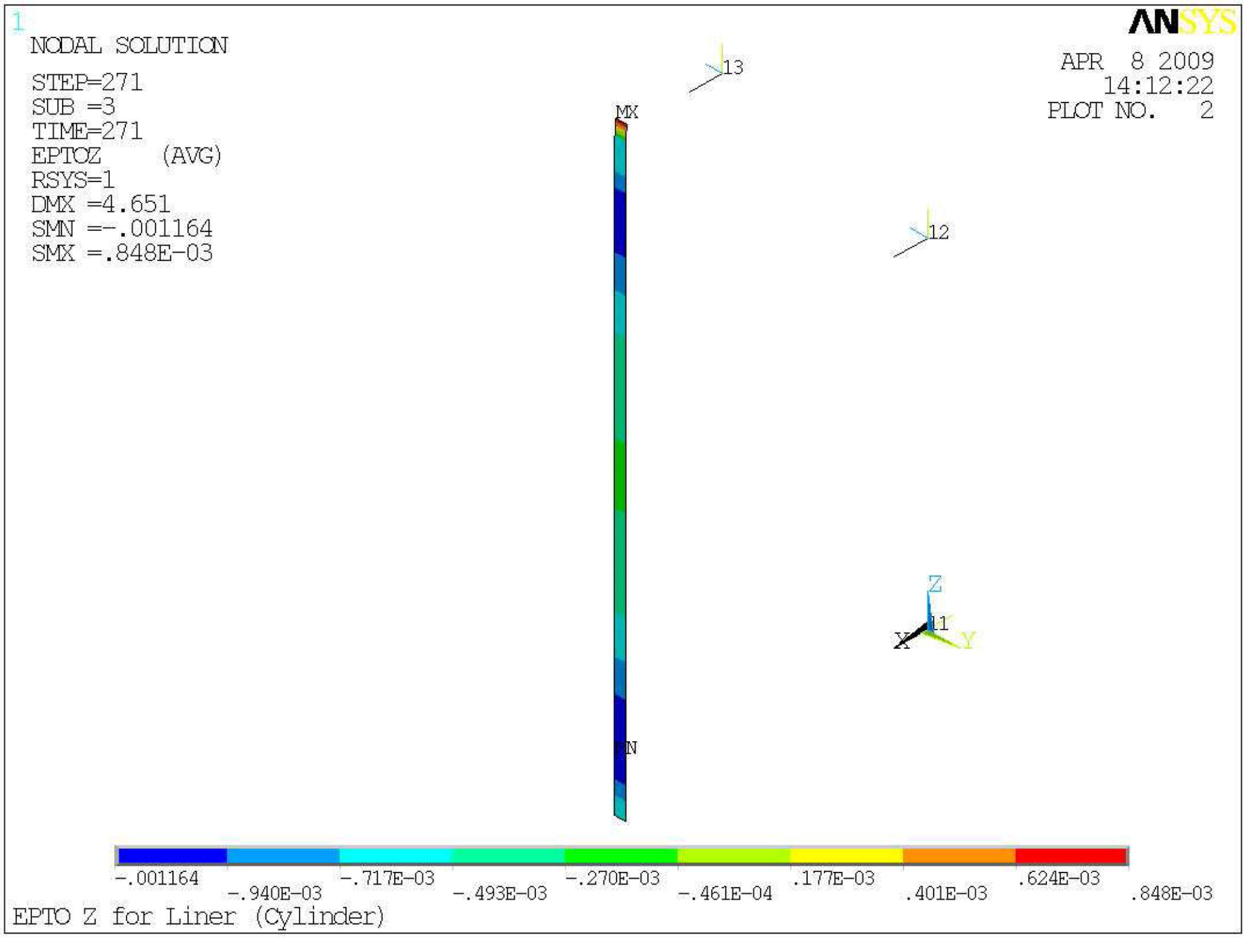


Figure 19-307-9—Tendon Hoop Strain (ε_y) in Cylinder

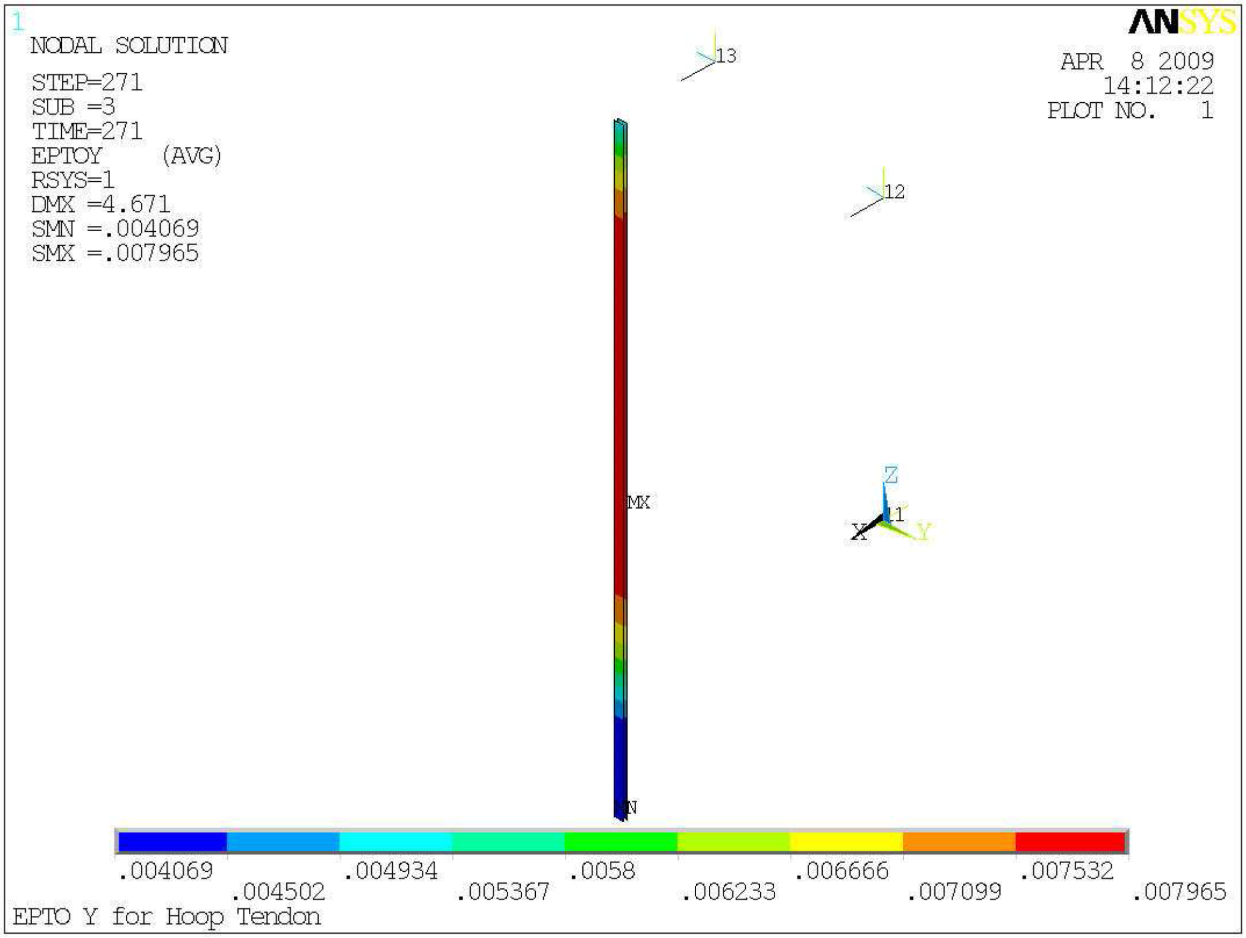


Figure 19-307-10—Tendon Meridional Strain (ε_z) in Cylinder