Comparison of Reg. Guide 1.99 Fluence Attenuation Methods

Eric N. Jones, Consultant

TransWare Enterprises Inc.

1565 Mediterranean Dr.

Sycamore, IL 60178, USA

ABSTRACT

U.S. Regulatory Guide 1.99 (RG 1.99) Revision 2 [1] provides for the use of two substantially different methods for determining through-wall fluence in nuclear reactor pressure vessels. One method is a generic attenuation curve based on a simplistic exponential decay equation. Partly due to its simplicity of application, the generic attenuation method is predominantly used for licensing calculations. However, it has a limitation that at increasing distances away from the core beltline, the generic method becomes increasing less accurate because it can not account for neutron streaming effects in the cavity region surrounding the pressure vessel. The other attenuation method is based on a displacement per atom (dpa) calculation specific to the reactor vessel structure. The dpa method provides a more accurate representation of fluence attenuation through the RPV wall at all elevations of the pressure vessel because it does account for neutron streaming in the cavity region. A requirement for using the dpa method, however, is an accurate flux solution through the RPV wall. This requirement has limited the use of traditional transport methods, such as Discrete Ordinates, that are limited by their treatment of cavity regions (i.e., air) outside the pressure vessel wall. TransWare Enterprises, under sponsorship of EPRI and BWRVIP, has developed an advanced three-dimensional transport methodology capable of producing fully converged flux solutions throughout the entire reactor system, including the cavity region and primary shield structures. This methodology provides an accurate and reliable determination of through-wall fluence in BWR and PWR pressure vessels; thus allowing the dpa method to be implemented with high reliability. Using this advanced 3-D methodology, this paper presents comparisons of the generic and dpa attenuation methods at critical locations in both BWR and PWR pressure vessel walls.

KEYWORDS:

RPV fluence, fluence attenuation methods, P-T curves, Regulatory Guide 1.99, vessel fluence, dpa

I. INTRODUCTION

One of the major components of plant licensing operations is the generation of pressure vs. temperature (P-T) limit curves for the reactor pressure vessel (RPV). These P-T curves determine the vessel preheat conditions for starting the reactor from a cold shutdown, ultimately affecting the amount of time needed to bring a reactor on-line. These curves are based on adjusted reference temperature (ART) calculations derived from through-wall fluence calculations for the various welds, plates, and forgings in the RPV beltline. These limits, in turn, dictate the operating margins that must be maintained in order to meet the safety requirements of 10 CFR 50 [2] as they relate to fracture toughness caused by neutron embrittlement. Since through-wall fluence is not a measurable quantity, and is a primary factor in deriving P-T limit curves, the computational methods used to calculate through-wall fluence play a critical role in the plant licensing process.

RPV through-wall fluence can be calculated in a number of different ways; however, RG 1.99 allows for only two methods to be used. RG 1.99 makes no mention as to the merits or applicability of one method over the other, leaving it to the licensee to pick the most applicable method. The first of the two prescribed methods is the generic attenuation equation, which determines the fluence a distance *x* through the RPV (in inches), written as:

$$f = f_{surf}(e^{-0.24x}) \qquad (1)$$

where:

f = fluence at a depth *x* inches through the RPV wall, and

f_{surf} = RPV inner wetted surface fluence, E>1.0 MeV.

The other method is based on the ratio of dpa at the desired depth to dpa at the surface. Following a similar format to equation (1), the dpa method can be written as:

$$f = f_{surf}\left(\frac{dpa_x}{dpa_{surf}}\right) \qquad (2)$$

where:

dpa_x = dpa at a distance x through the RPV wall, and

dpa_{surf} = dpa at the RPV inner wetted surface.

At the time RG 1.99 was written, traditional methods employing 3-D synthesis operations were not capable of determining accurate fluence beyond the core beltline in RPV materials. These methods are inherently limited at achieving convergence beyond the RPV wall inner surface. However, modern particle transport codes, capable of true 3-D modeling well beyond the core beltline region, yield detailed, converged and accurate accounts of fluence throughout the entire reactor system, including the primary shield wall.

Modern calculations show that many factors can affect the RPV through-wall fluence in ways that are not accurately accounted for by the generic attenuation method. Although the generic attenuation method is accurate near the core mid-plane where the peak RPV fluence is expected to occur, it does not properly reflect the influence of neutron cavity streaming. Neutron cavity streaming contributes a proportionally significant amount of fluence to the outer RPV wall surface at the beltline transition elevations. As a result, the generic attenuation equation can be increasingly non-conservative at increasing distances away from the core beltline.

II. ANALYSIS

In order to determine the significance of the difference between the two methods on the U. S. nuclear fleet, five different U. S. nuclear power plant designs were evaluated. These designs included three different boiling water reactor (BWR) designs and two pressurized water reactor

(PWR) designs. Evaluations were performed in full three-dimensional geometry using actual reactor operating data and fuel assembly designs. Fluences for four of the reactors were projected to the end of the reactors' extended operating lives (60-year). Fluence for one reactor, the PWR with 157 fuel assemblies, was based on one cycle of operation.

Three representative elevations were chosen for each plant type in order to determine the general trending of the generic and dpa attenuation methods through the RPV beltline region. Two elevations represented the upper and lower beltline transition elevations where the fluence in the RPV exceeds 1.0E+17 n/cm^2, and one elevation represented the core mid-plane.

The RAMA Fluence Methodology [3], hereafter referred to as RAMA, was used to construct detailed reactor fluence models of the five subject plants and to perform the transport and fluence calculations. RAMA uses a three-dimensional deterministic solution of the steady-state Boltzmann transport equation. The solution methodology is based upon the method of characteristics coupled with arbitrary geometry modeling capability. The standard RAMA nuclear data library is based upon the BUGLE-96 [4] cross section data set with additional nuclide data from VITAMIN-B6 [5]. Anisotropic scattering is provided using traditional Legendre scattering moments. The individual approximations are based upon the maximum order of expansion available for each nuclide in the RAMA nuclear data library, P_5 for actinide and zirconium nuclides and P_7 for all others.

RAMA calculates a weighted fission spectrum, based on the relative contributions of the fuel isotopes, for each fission region in the reactor core. The fission spectra for ^{235}U, ^{238}U, ^{239}Pu, ^{240}Pu, ^{241}Pu, and ^{242}Pu that are used in the RAMA transport calculations are taken directly from the latest release of the BUGLE-96 data library. RAMA is approved by the Nuclear Regulatory Commission (NRC) for use in determining fluence in the RPV [6] and reactor internals [7] in accordance with U.S. Regulatory Guide 1.190 [8].

Core simulator data is used to model the reactor core operating history. The core simulator data is input in the form of state-point data files. State-point files include three-dimensional data arrays that describe core power distributions, fuel exposure distributions, fuel materials, water densities, and soluble boron concentrations (in PWRs) in the reactor core. A separate neutron transport calculation is performed for each of the available state points. The calculated neutron flux for each state point is then combined with the appropriate power history data in order to provide an accurate accounting of the fast neutron fluence for the reactor components and surveillance capsules.

The beltline elevations were determined based on the existing fluence evaluations performed by TransWare for the subject plants. Through-wall RPV fluence was then calculated at the beltline limits and the core mid-plane elevations using Equations (1) and (2) in 1-cm increments through the vessel wall. Figures 1-5 illustrate the fluence profiles for the five subject plants.

III. OBSERVATIONS

In general, it was determined for the subject BWR designs that the jet pump instrumentation nozzles (N16) and LPCI nozzles (N17), if present, lie within the RPV beltline elevations for the BWR configurations. The recirculation outlet and inlet nozzles (N1 and N2) generally lie outside of the RPV beltline region. In the PWR configurations, the RPV beltline region extended too close to the inlet and outlet nozzles to make a clear assessment as to whether they would be included or excluded for most units. Several factors, including power uprates, cycle lengths, fuel enrichments and plant life extension parameters will play a deciding role in the inclusion of nozzles in the beltline region for PWRs. For all plant types, the RPV beltline region assessments were made based on 60-year extended operating lives.

It was observed that for all but the smallest BWR, the generic attenuation method produced results that agreed well with the dpa method at the core mid-plane elevation, which is where the fluence attenuations are primarily used in evaluating peak plate through-wall fluence. The minor differences shown between the generic and dpa methods illustrate the significance of performing reactor-specific calculations. Since the generic method is meant to apply to all reactor designs, it performs best for reactor designs of moderate size and power density. Designs at either end of the size or power density spectrum are more affected, as illustrated by the BWR designs with 368 and 764 bundles, that don't compare as well as the 560-bundle design.

However, for all reactor configurations, the generic equation produced significantly non-conservative results at the boundaries of the RPV beltline region. The amount of non-conservatism varied by plant configuration, but was generally more pronounced for smaller BWR's and for both PWR reactor types. In the most extreme example, the dpa attenuation exceeded the fluence predicted by the generic attenuation by almost 700 % at the outer surface of the vessel at the upper beltline elevation in the PWR with 193 fuel assemblies.

The reason for the disparity between the two attenuation methods is the result of the streaming of neutrons in the cavity region. The fast neutrons that escape the RPV redistribute themselves throughout the cavity region, with little or no loss of energy. These fast neutrons then re-enter the vessel wall from the outer surface at elevations well above and below the peak elevation. Since the nitrogen in the cavity has virtually no moderating effect on the neutrons, the fast fluence on the outer surface of the RPV will eventually exceed the fluence on the inner surface. Due to the fact that the generic attenuation calculation can not account for neutron streaming in the cavity region, the results of this method can potentially be grossly inaccurate at predicting the through-wall fluence at nozzle forgings in both the BWR and PWR reactor fleet.

IV. CONCLUSION

This paper shows that the dpa attenuation method described in RG 1.99, which can account for neutron streaming outside the pressure vessel wall, provides a more accurate representation of the through-wall fluence for RPV embrittlement evaluations. Based on comparisons to the dpa method, it is further shown that the generic attenuation equation becomes increasingly non-conservative with increasing distance from the reactor core beltline for all reactor types and configurations. The trending curve through the pressure vessel wall shows different responses for BWRs versus PWRs, most likely due to differences in pressure vessel wall thicknesses and geometric configurations of the cavity and primary shield structures of the two reactor types. The degree of non-conservatism shown for the different classes of reactor types is also shown to vary appreciably, most likely due to the different reactor core configurations (i.e., number of fuel assemblies) and pressure vessel diameters. Based on the merits of these comparisons, selecting the dpa method for RPV embrittlement evaluations seems obvious; however, implementing an accurate dpa method requires an accurate determination of flux and fluence outside the RPV wall, which in turn requires a transport methodology capable of producing converged, accurate and reliable results for the reactor system. TransWare Enterprises, under sponsorship of EPRI and BWRVIP, developed the RAMA Fluence Methodology which has been demonstrated to generate accurate 3-D flux and fluence profiles throughout the reactor system without adjustments and with zero bias. Additional measurement data on the outer surface of the RPV at elevations comparable to the beltline transition region would be beneficial in further benchmarking the transport codes and attenuation methods.

REFERENCES

[1] Nuclear Regulatory Commission Regulatory Guide 1.99, Revision 2, 1988: Radiation Embrittlement of Reactor Vessel Materials.

[2] Code of Federal Regulations, Title 10, Part 50, Appendix G, Jan. 31, 2008: Fracture Toughness Requirements.

[3] BWRVIP-126, Revision 2: BWR Vessel and Internals Project, RAMA Fluence Methodology Software, Version 1.2, EPRI, Palo Alto, CA: 2010. 1020240.

[4] BUGLE-96: Coupled 47 Neutron, 20 Gamma-Ray Group Cross Section Library Derived from ENDF/B-VI for LWR Shielding and Pressure Vessel Dosimetry Applications, RSICC Data Library Collection, DL2C-185, March 1996.

[5] VITAMIN-B6: A Fine-Group Cross Section Library Based on ENDF/B-VI Release 3 for Radiation Transport Applications, RSICC Data Library Collection, DLC-184, December 1996.

[6] Letter from William H. Bateman (U. S. NRC) to Bill Eaton (BWRVIP), "Safety Evaluation of Proprietary EPRI Reports BWRVIP-114, -115, -117, and -121 and TWE-PSE-001-R- 001," dated May 13, 2005.

[7] Letter from Matthew A. Mitchell (U. S. NRC) to Rick Libra (BWRVIP), "Safety Evaluation of Proprietary EPRI Report BWR Vessel and Internals Project, Evaluation of Susquehanna Unit 2 Top Guide and Core Shroud Material Samples Using RAMA Fluence Methodology (BWRVIP-145)," dated February 7, 2008.

[8] Nuclear Regulatory Commission Regulatory Guide 1.190, March 2001: Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence.

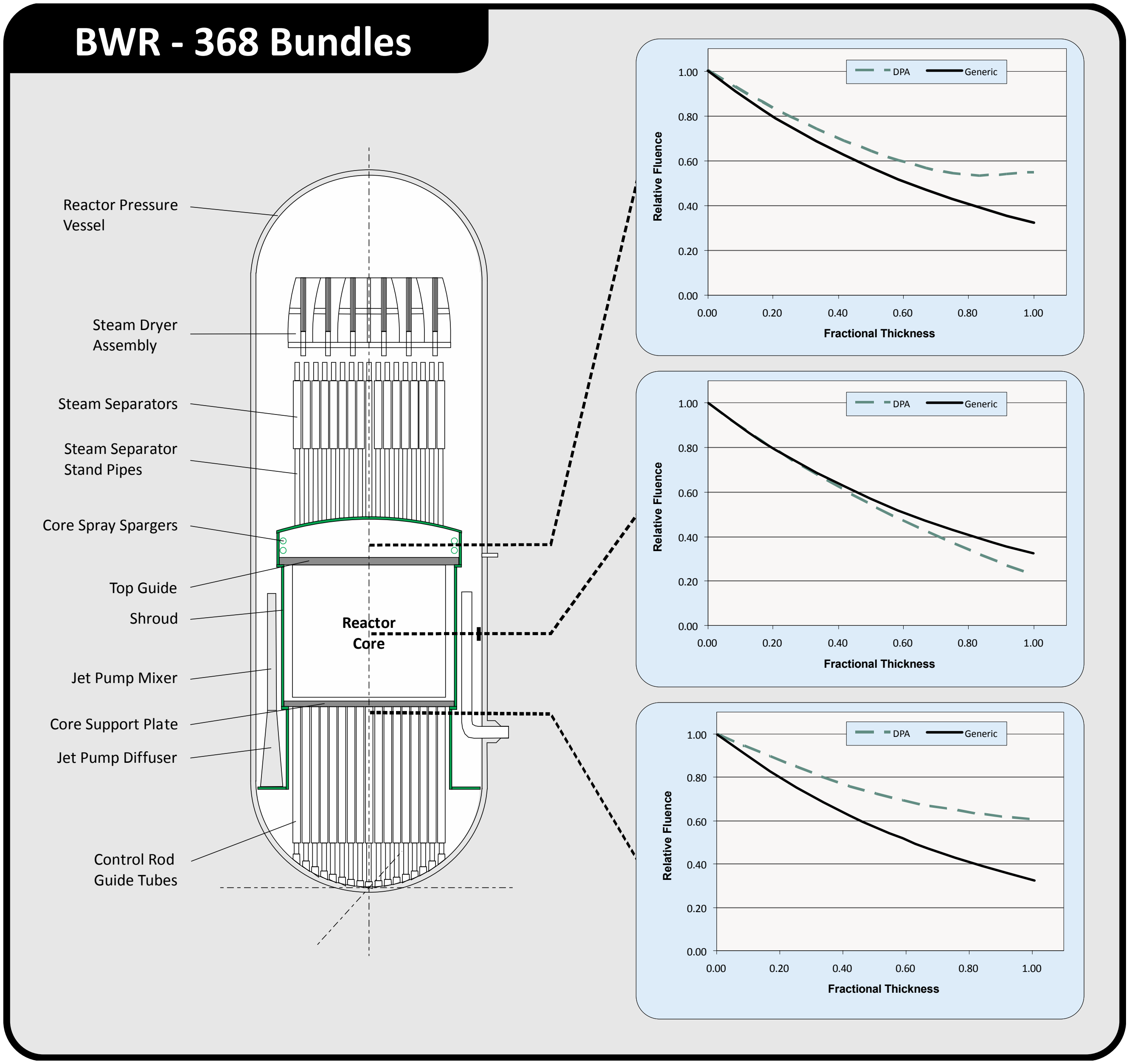

Fig. 1 – BWR with 368 Fuel Assemblies

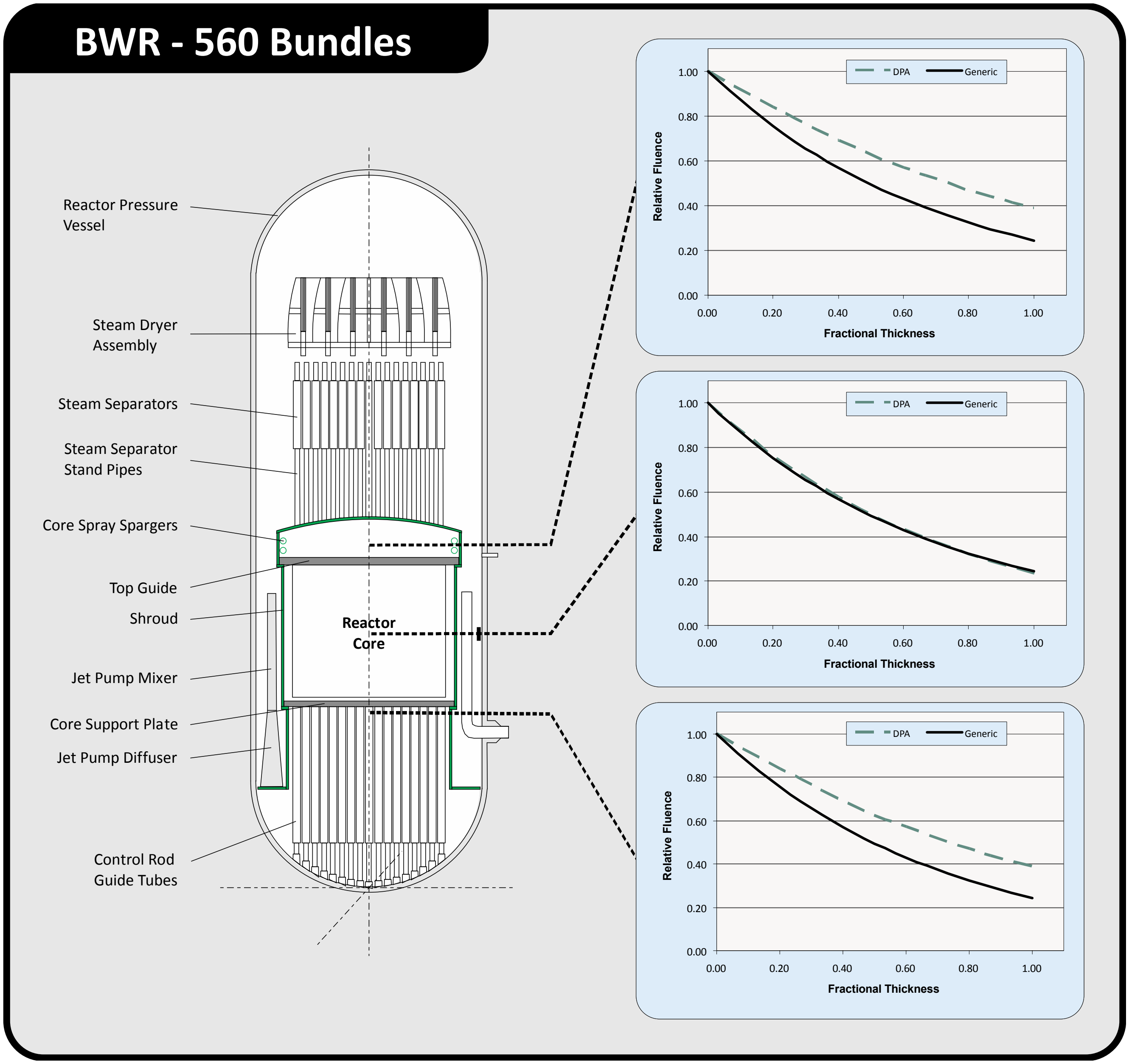

Fig. 2 – BWR with 560 Fuel Assemblies

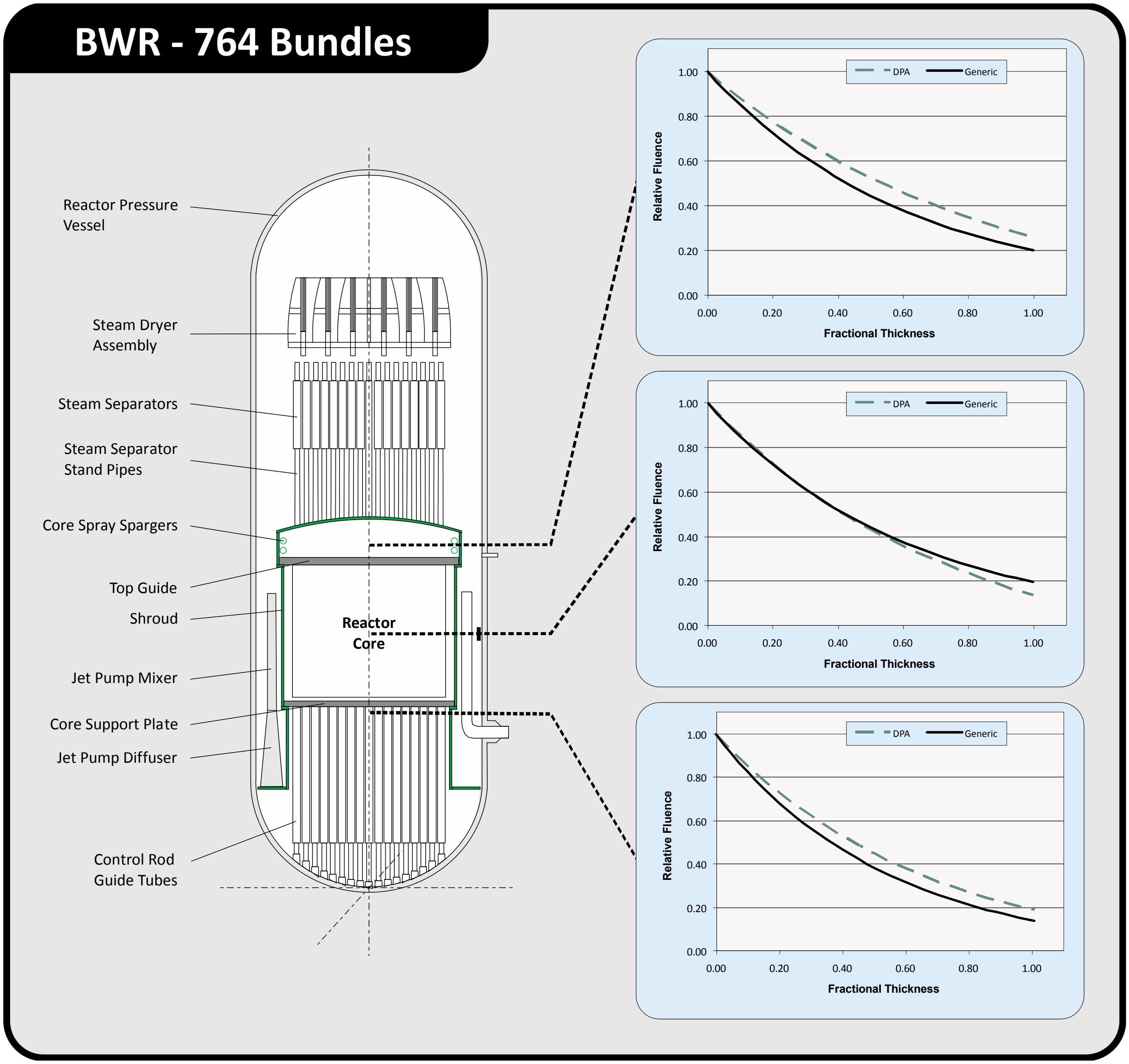

Fig. 3 – BWR with 764 Fuel Assemblies

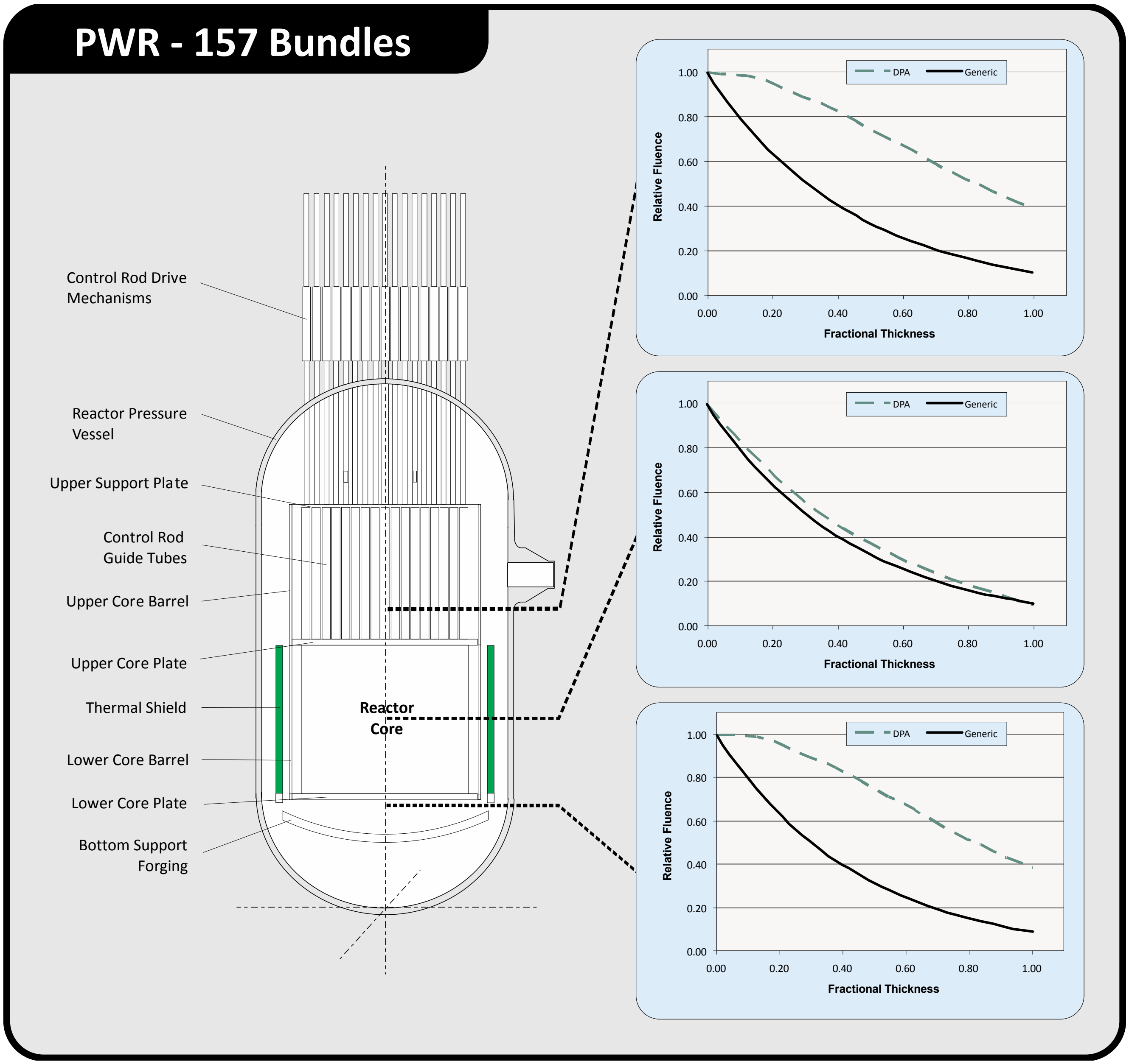

Fig. 4 – PWR with 157 Fuel Assemblies

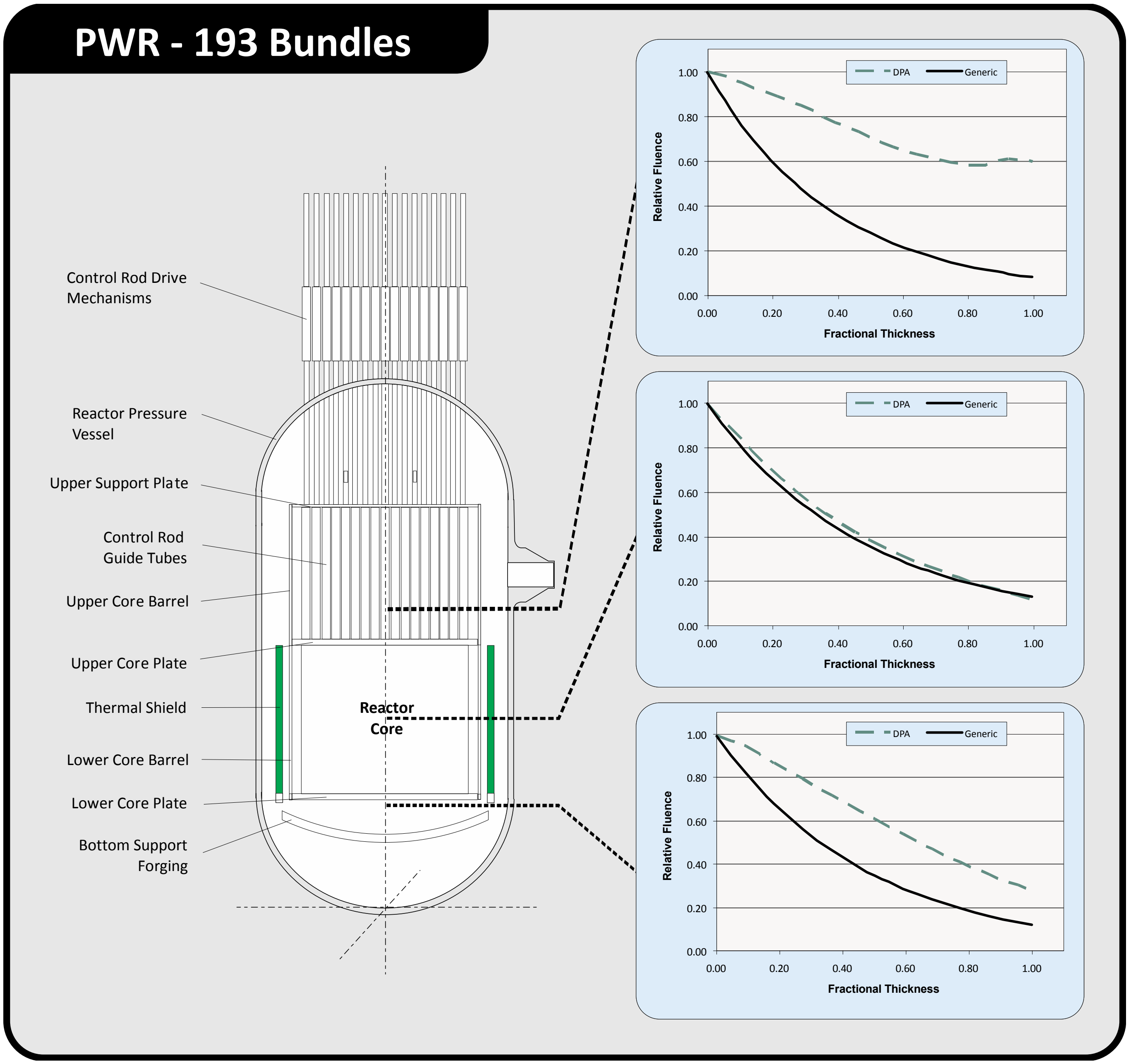

Fig. 5 – PWR with 193 Fuel Assemblies