From Maximum Moderation to Engineering Selection: Multi-Physics Screening of YH_x and BeO in a Skewer-Type Fuel for a 140 MWe Soluble-Boron-Free SMR

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1. Introduction

A strong solid moderator is not necessarily the most suitable insert at the center of a water-cooled UO_2 pellet. Moderator effectiveness must be considered together with displaced fuel volume, thermal resistance, fabrication tolerances, contact loads, irradiation evolution, and hydrogen transport. This study compares Zr-Nb-encapsulated yttrium hydride (YH_x) and bare beryllium oxide (BeO) in a Skewer-Type Fuel (STF), in which an axial center member passes through annular UO_2 pellets.

The design context is a conceptual 450 MWth / 140 MWe natural-circulation, soluble-boron-free SMR. This paper is a material-selection and analytical screening study, not a demonstration of fuel-rod integrity. It separates calculated geometric quantities from literature evidence and from qualification work that remains to be performed. The comparison preserves the potential value of YH_x while identifying why BeO is provisionally retained.

YH_x fabrication, thermophysical characterization, and reactor-physics studies provide a basis for considering a hydride insert [1–3]. BeO properties and irradiation limitations are documented separately [4]. Neither literature base qualifies the present insert/wire/pellet/cladding assembly. In particular, a positive nominal clearance and an intended wire-support function are not evidence of acceptable production yield or in-reactor stress.

2. Controlled Comparison and Methods

2.1 Geometry and intended architecture

The common design basis comprises 69 assemblies, 236 fuel rods per assembly, a 2.40 m active stack, and 5.00 wt.% U-235. The UO_2 pellet outer diameter is 8.192 mm; cladding outer diameter and thickness are 9.50 and 0.57 mm, giving an 8.36 mm inner diameter. Approximately 240 pellets form each active stack.

The BeO reference has a 2.640 mm bore and a 2.590 mm bare center rod. Two opposed round Zr-Nb interface wires are intended to provide centering and limited local heat transfer; their ends nominally reach the cladding. Figure 1 depicts this unqualified contact architecture. Wire clocking between interfaces is not credited with eliminating local stress concentrations.

The $YH_{1.90}$ fuel-recovery case uses a 1.500 mm capsule outer diameter, 0.150 mm wall, 1.200 mm inner diameter, 1.180 mm hydride core, and 1.550 mm pellet bore. The same-envelope sensitivity uses a 2.590 mm capsule and a 2.270 mm hydride core in the 2.640 mm bore. The assumed capsule thickness is a screening input, not a pressure-boundary design approval.

2.2 Reproducible screening equations

For pellet diameter D_p, bore diameter D_b, and a fixed groove-volume allowance $\varepsilon_g = 0.0009234$, the geometric uranium displacement and inventory are

$$R_U = (D_b/D_p)^2 + \varepsilon_g;\ \ M_U = M_{ref}(1 - R_U) \quad (1)$$

where $M_{ref} = 19.0224$ tU and $D_p = 8.192$ mm. For the solid reference, both bore and groove allowance are zero. The allowance is held fixed only for comparison; any support redesign requires a new volume calculation.

At equal rod linear power, constant UO_2 conductivity, equal outer-surface temperature, uniform heat generation within each fuel region, and an adiabatic bore, the annular-to-solid maximum radial temperature-rise ratio is

$$\Delta T_{ann}/\Delta T_{solid} = 1 + f\ \ln(f)/(1 - f) \quad (2)$$

with $f = (D_b/D_p)^2$. Equal linear power, rather than equal volumetric heat generation between geometries, is essential to this expression. This shape-only result does not credit BeO conduction or wire heat bypass and does not predict the irradiated fuel temperature.

For a deliberately simple one-group comparison, define the insert-volume ratio $v = (D_m/2.590)^2$, where D_m is the moderator diameter in millimeters. The relative integrated slowing-down and absorption screens are

$$S = [K_m/K_{BeO}]v;\ \ A = [\Sigma_{a,m}/\Sigma_{a,BeO}]v \quad (3)$$

The retained nominal inputs are $K = \xi\Sigma_s = 4.428$ / 0.151 cm^{-1} and $\Sigma_a = 0.05426$ / 0.0005646 cm^{-1} for $YH_{1.90}$ / BeO. They are assumed screening constants, not a newly evaluated nuclear-data set. Capsule and wire absorption are excluded. Multigroup transport with temperature-dependent thermal scattering, depletion, and feedback is required before inferring enrichment, cycle length, or net uranium savings [3].

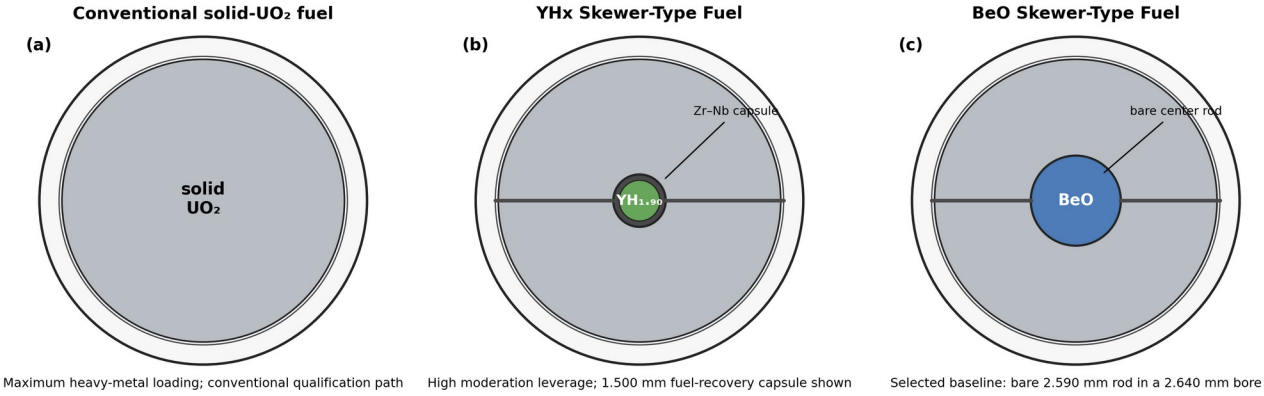


Fig. 1. Conceptual pellet-interface cross-sections: solid UO_2, $YH_{1.90}$ STF, and BeO STF. Intended wire-to-cladding contacts are unqualified. Small gaps and wire-tip radii are not resolved; Table 1 and Section 3 define the numerical clearances.

Table 1. Recalculated nominal geometry and analytical screens; not in-reactor performance predictions.

Architecture	**Bore / insert / hydride diameter (mm)**	**Core U (tU)**	**Ideal ΔT reduction**	**S / A (relative to BeO)**
Solid UO_2	No bore / no insert	19.022	0%	— / —
$YH_{1.90}$, fuel recovery	1.550 / 1.500 / 1.180	18.324	12.36%	6.09 / 19.95
$YH_{1.90}$, same envelope	2.640 / 2.590 / 2.270	17.029	26.25%	22.53 / 73.82
BeO, reference	2.640 / 2.590 / —	17.029	26.25%	1.00 / 1.00

Insert diameter denotes capsule outer diameter for YH_x and bare-rod diameter for BeO. Values use Eqs. (1)–(3); rounding follows recomputation from the stated constants.

2.3 Interpretation and limits of the comparison

The nominal BeO geometry displaces 10.4779% of the reference uranium inventory, or 1.993 tU. The 1.500 mm YH_x capsule case displaces 3.6723%, restoring 1.295 tU (7.60%) relative to BeO. At 450 MWth, average specific powers are 23.66, 24.56, and 26.43 kW/kgU for solid UO_2, the YH_x fuel-recovery case, and BeO, respectively.

These are mass and power identities, not fuel-cycle savings. The YH_x case contains more uranium than BeO while retaining a larger assumed moderation screen. Its neutron-economy benefit must be weighed against absorption, spectral changes, leakage, control worth, and feedback. No enrichment reduction or longer cycle is calculated here.

The 26.25% BeO-case and 12.36% YH_x-case thermal values arise from bore geometry under Eq. (2), not from a measured insert cooling effect. The same-envelope YH_x case gives the same ideal shape benefit as BeO; its internal gap, capsule wall, and external contact introduce additional thermal interfaces that the ideal equation does not resolve.

Hydride manufacture and thermophysical data justify further study [1,2]. Neutron-irradiation work also reports encouraging phase and structural stability over its tested conditions, while warning that dimensional and mass measurements alone can misrepresent hydrogen retention [5]. Those observations should not be extended to the lifetime, stress, hydrogen inventory, or coolant environment of this fuel design without qualification.

A chemically compatible oxide interlayer is not automatically a hermetic hydrogen barrier. Compatibility testing [6] and desorption characterization [7] address different questions from permeation, end-closure leakage, irradiation damage, and pressure loading in a slender capsule. A barrier option is therefore not credited in Table 1 or in the material-selection conclusion.

BeO avoids the hydride moderator's hydrogen reservoir and the sealed inner capsule. This is a reduction in coupled design variables, not proof that BeO or its Zr-Nb supports are safe. BeO fracture, irradiation-induced property changes, swelling, helium generation, occupational controls, and fragment retention remain explicit qualification subjects [4]. Sections 3–5 translate these unresolved issues into measurable requirements.

3. Manufacturing and Assembly Assessment

3.1 Radial clearance and tolerance budget

The nominal 25 μm clearance is radial, not diametral: $g_{r,0}$ = (2.640 − 2.590)/2 = 0.025 mm. The previous drawing assumptions, D_b = 2.640 ± 0.005 mm and D_s = 2.590 +0.000/−0.005 mm, give a dimensional radial-clearance range of 22.5–30.0 μm. They are not demonstrated commercial production capabilities.

A conservative available-clearance budget is

$$g_{r,min} = (D_{b,min} - D_{s,max})/2 - e_{eff} \quad (4)$$

where e_{eff} is a radial allowance for relative eccentricity, bow, stack misalignment, and residual surface/debris effects not already included in the measured diameter envelopes. The same form error must not be counted twice. Table 2 varies this allowance; the values are assumptions for sensitivity, not manufacturing measurements.

Table 2. Available radial gap at the previous adverse diameter limits.

Assumed e_eff (μm)	g_r,min (μm)
0	22.5
10	12.5
20	2.5
25	−2.5

A negative value represents geometric interference in this budget, not a calculated chipping force. A positive value alone does not exclude edge contact during insertion through a long pellet stack.

3.2 Required assembly evidence

Manufacturing release requires supplier-specific distributions and inspection limits for bore/rod diameter, roundness, straightness, concentricity, and pellet-edge condition. Controlled entry chamfers or radii, clean handling, guided insertion, and an insertion-force stop criterion must be developed without assuming that these measures eliminate chipping.

Representative full-length assembly trials must record force versus insertion travel, pre/post-assembly edge damage, loose particles, yield, and repeatability across production lots. Acceptance limits for damage and force must be established before testing. No assembly tests or process-capability results are reported here; therefore, 25 μm is retained only as a comparison input, not released as a production clearance.

3.3 Clearance enlargement is a design trade

At fixed 2.590 mm BeO diameter, larger nominal bores improve geometric assembly space but remove additional UO_2 and change thermal/contact behavior. Table 3 recalculates only the geometric consequences; it does not select a replacement clearance.

Table 3. Illustrative bore sensitivities at fixed BeO diameter; groove allowance held constant.

Bore (mm)	Radial gap (μm)	Core U (tU)	Ideal ΔT reduction
2.640	25	17.029	26.25%
2.690	50	16.954	26.92%
2.740	75	16.877	27.59%

The larger ideal shape benefit does not establish a lower operating temperature: gap conductance, centering, contact, neutronics, and fragment retention must all be reassessed. An alternative is a smaller insert at fixed bore, which changes moderator inventory rather than UO_2 inventory.

4. Coupled Thermomechanical Assessment

4.1 Contact onset is not cladding strain

At a specified axial/azimuthal location, radial separation can be expressed kinematically as

$$g_r(t) = g_{r,0} + u_b(t) - u_s(t) - e(t) \quad (5)$$

where u_b and u_s are bore and insert-surface radial displacements from their reference positions, and e is their relative offset. Contact begins where the free-gap prediction reaches zero; subsequent loads require a coupled contact solution. The displacements must include thermal expansion, fuel densification/swelling, cracking/relocation, irradiation creep/growth, and capsule deformation as applicable.

For illustration only, closing a 25 μm gap against a fixed bore would require 25/1295 = 1.93% outward radial growth of the BeO rod. The YH_x capsule's internal radial gap is 10 μm, corresponding to 10/590 = 1.69% hydride-radius growth against a fixed capsule. These are purely geometric closure scales, not predicted swelling, allowable strain, or actual cladding deformation.

The UO_2 bore cannot be assumed to expand freely after pellet/cladding or insert contact. Component temperatures must be solved separately; applying a single coolant temperature to every material would not bound differential expansion. Axial stack growth, end restraints, and capsule/plenum free volume also enter the load path.

4.2 Model inputs and acceptance outputs

A rod-scale fuel-performance model should establish temperature, fission-gas pressure, swelling, and axial/radial deformation histories. A three-dimensional local model must then resolve the pellet groove, finite-radius wire end, cladding wall, and insert/capsule contacts. Multidimensional fuel-performance approaches provide a methodological basis [8], but no BISON or finite-element calculation is reported for this design.

Required inputs include alloy grade and heat treatment, irradiated constitutive data, hydrogen-dependent ductility, wire diameter/end radius, preload, friction, contact conductance, as-built tolerances, power/temperature histories, and external/internal pressures. Until these inputs are fixed, a numerical peak cladding stress or strain would be unsupported.

Evaluation must report local contact pressure, principal and hoop stresses, equivalent plastic strain, cyclic strain, residual wall thickness after fretting, wire deformation/retained force, and gas-path continuity. Limits must be justified for the actual alloy, irradiation state, temperature, hydrogen content, and loading mode. Generic membrane-stress or single uniform-strain limits cannot alone demonstrate the acceptability of a wire-tip contact. SRP 4.2 is used as a fuel-integrity assessment reference, not as a claim of Korean licensing compliance [9].

4.3 Wire contact and loss of support

A round wire meeting the curved cladding wall can produce a small contact area and high local stress. Rotating the wire orientation by 90° between interfaces changes contact locations but does not bound the peak stress at any one location. Rounded or broadened contact ends and a compliant load path are redesign candidates only; their absorption, geometry, and fretting consequences require reassessment.

The minimum failure-case set is intact support, preload relaxation/irradiation creep, plastic wire deformation, one-wire fracture, and correlated support loss at adjacent interfaces. The degraded configurations must be assessed for insert eccentricity/bowing, pellet displacement, changed contact loads, wire/BeO fragments, fission-gas communication, and cladding damage during power changes and cooldown.

No primary heat-removal or safety-retention credit is assigned to the wires in this screening. Removing wire thermal conductance is a required sensitivity, not proof of acceptable temperatures. Independent retention or another demonstrated fail-safe arrangement is a design requirement; it is not shown as a completed feature in Fig. 1. Failure to qualify the local contacts or the support-loss configuration requires redesign, including reconsideration of direct wire-to-cladding contact.

5. YH_x Hydrogen and Capsule Integrity

5.1 Inventory-conserving pressure assessment

Hydrogen release, capsule pressure, permeation, and wire uptake must be considered together [7]. For n_Y moles of yttrium, a stoichiometry decrease from x_0 to x corresponds to a released hydrogen inventory of $n_Y(x_0 - x)/2$ moles on an H_2-equivalent basis. Release does not imply that all hydrogen remains as free molecular gas.

$$n_{cap} = n_0 + n_{rel} - n_{out} - n_{uptake} \quad (6)$$

$$P_{cap} = Zn_{cap}RT/V_{free};\ \Delta P = P_{cap} - P_{rod} \quad (7)$$

Here n_{cap} denotes free capsule-gas moles, n_0 the initial gas inventory, n_{rel} the net gas-equivalent release from the hydride, n_{out} net loss by leakage/permeation, and n_{uptake} net absorption into capsule materials; Z is a gas-model factor. A multicomponent equation of state and chemical equilibrium may be needed. V_{free} changes with temperature, swelling, and capsule deformation.

Hydrogen leaving the capsule must be accounted for in the surrounding fuel-rod gas space and in uptake by the wire/cladding. A lower capsule pressure from leakage cannot be claimed as an integrity benefit without that downstream balance. End-closure strength, external-pressure collapse, internal overpressure, local contact, and thermal cycling must all be evaluated. No pressure number is assigned without a qualified release law, free volume, and transport data.

5.2 Embrittlement and qualification gate

Tests on zirconium-alloy cladding show that prior hydrogen pickup affects embrittlement under LOCA-relevant conditions [10]. Hydrogen-redistribution modeling also identifies temperature/stress gradients as important to local hydride accumulation [11]. These findings motivate, but do not quantify, the risk in the present slender Zr-Nb wire; cladding data cannot be transferred directly to its geometry and fabrication state.

The required wire program must cover the actual alloy/heat treatment and surface condition, measured hydrogen uptake, hydride distribution/orientation after cooldown, retained ductility, fracture resistance, cyclic contact/fretting, and delayed hydride cracking where applicable. Capsule-defect and barrier-degradation cases must be included. Total hydrogen content alone is not an adequate surrogate for local mechanical performance.

YH_x remains outside the qualified design baseline until capsule retention/transport and hydrogen-exposed wire performance are demonstrated together. BeO removes this additional moderator-derived hydrogen source, but does not make zirconium-alloy components immune to hydrogen-related degradation from other sources.

6. Conditional Selection and Validation

BeO is retained as the provisional material reference, not as a qualified fuel system. It preserves the larger nominal annular geometry benefit and avoids a hydride-bearing inner capsule. Its occupational hazards, brittleness, irradiation-induced conductivity changes, swelling/helium generation, and fragment retention remain mandatory development issues [4]. Selecting BeO does not resolve the common wire/contact failure modes.

YH_x is retained as a research alternative rather than rejected as intrinsically unsuitable. Its strong assumed moderation screen can be used to explore a smaller center region and higher heavy-metal loading. Replacement of BeO requires both acceptable neutron-economy/feedback results and manufacturing, thermal, capsule, and hydrogen-exposed mechanical evidence. A benefit in one metric cannot compensate for failure of a fuel-integrity criterion.

The proposed validation sequence is: supplier-supported tolerances and full-length assembly trials; out-of-pile thermal/contact and support-loss tests; verified rod-scale and three-dimensional local analyses; capsule/wire and BeO irradiation with post-irradiation examination; and lead-test rod/assembly evidence. Neutronic work must use the as-built geometry and qualified H/Y envelope, including depletion, control worth, power peaking, and temperature/void feedback.

The program must define pass/fail criteria before testing and carry measurement/model uncertainty into the acceptance decision. Until these gates are passed, a conventional solid-UO_2 design remains a reference/fallback option, itself subject to qualification for the proposed SMR. No safety, cycle-length, enrichment, or licensing credit is taken from this conceptual comparison.

7. Conclusions

The recalculated nominal comparison gives 17.029 tU and a 26.25% ideal shape-only temperature-rise reduction for BeO STF. The YH_x fuel-recovery case contains 18.324 tU and retains a one-group slowing-down screen of 6.09 relative to BeO, but this does not establish net uranium savings.

The previous 25 μm radial gap gives only 22.5 μm at the adverse diameter limits before relative-offset allowances. Manufacturing feasibility therefore requires production-specific metrology and full-length assembly evidence. Larger-clearance cases have been quantified as sensitivities, not adopted as validated improvements.

Coupled deformation, local wire contact, support loss, and moderator-derived hydrogen are unresolved integrity issues. Their governing variables, required failure cases, and acceptance outputs are now explicit. BeO is a conditional material choice; direct wire-to-cladding contacts and the YH_x capsule remain subject to redesign or rejection if qualification cannot demonstrate acceptable performance.

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