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High-burnup Experiments in Reactivity Initiated Accidents (HERA)

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Changing the World's Energy Future

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ABSTRACT

HERA is a joint experimental program (JEEP) operating within the Nuclear Energy Agency's (NEA's) framework for irradiation experiments (FIDES). HERA is dedicated to the understanding of light water reactor (LWR) fuel performance at high burnup under reactivity-initiated accidents (RIA). In-pile RIA experiments have been performed on high burnup fuels (above 60 GWD/MTU) in the CABRI reactor in France, and the NSRR reactor in Japan. However, the majority of these experiments have taken place with heavily corroded Zircaloy claddings in test reactors with pulse widths that are more narrow (5ms – 20ms full-width-half-max (FWHM)) than what would be likely in a commercial LWR (30ms – 80ms FWHM). Heavy waterside corrosion and narrow pulse widths are both known to increase the vulnerability of LWR fuel to pellet cladding mechanical interaction (PCMI). The HERA proposal is designed to (1) quantify the impact of pulse width on fuel performance, offering new insight into the applicability of existing data, (2) generate new data on high burnup fuel under pulse conditions prototypic of LWRs (3) quantify the additional margin provided by modern cladding alloys to PCMI failure limits and (4) offer improved data for modelers using specially designed tests that eliminate key uncertainties in high-burnup fuel tests.

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1. BACKGROUND

During RIAs the fuel cladding can be breached by one of two mechanisms; pellet-cladding mechanical interaction or high temperature balloon and rupture. Pellet-cladding mechanical interaction (PCMI) failure occurs in the early phase of an RIA transient where the fuel pellet expands rapidly into the cladding prior to any meaningful heat transfer from the pellet to the cladding. This results in high stresses in the cladding resulting in crack propagation through the base Zircaloy metal and axially along the cladding tube. Failure by PCMI is a brittle failure mode and occurs at low cladding strains, approximately 2% hoop strain.

If the cladding survives the PCMI phase of the transient, it may still fail due to ballooning and rupture induced by extended exposure to high temperature. Thermal transport from the fuel pin to the coolant is significantly reduced if a transition to film boiling on the cladding surface occurs. This result in both a rapid increase in fuel temperature, which drives fission gas release, and a spike in cladding temperature, which results in loss of mechanical strength. If the internal pressure of the fuel rod is greater than the system pressure the cladding can inelastically deform and eventually rupture. Failure by swelling and rupture occurs at larger strains, generally greater than 5% hoop strain. While PCMI failures are generally seen as more limiting, the dominance of one failure mode over the other can be affected by both the cladding material conditions, and the transient evolution.[1]

Due to the complex multi-physics interactions, fuel performance in RIA is best studied in test reactors especially suited for replicating the rapid nuclear heating conditions in the fuel. In-pile RIA experiments have been performed on high burnup fuels (above 60 GWD/MTU) in the CABRI reactor in France, and the NSRR reactor in Japan.[2] However, the majority of these experiments have taken place with heavily corroded Zircaloy claddings in test reactors with pulse widths that are narrower (5ms – 20ms full-width-half-max (FWHM)) than what would be likely in a commercial LWR (30ms – 80ms FWHM). Heavy waterside corrosion and narrow pulse widths are both known to increase the vulnerability of LWR fuel to pellet cladding mechanical interaction (PCMI).[3]

In traditional Zircaloy claddings, thick oxide layers and zirconium hydrides form on the outer edge of the cladding where flaws and cracks can be easily nucleated. The severity of these oxide layers and cladding hydrides directly affect the fuel's vulnerability to PCMI failure. In Zircaloy-2 cladding used in Boiling Water Reactors (BWRs), radial hydrides forming near the cladding surface are known to greatly increase the vulnerability for the cladding to PCMI failure[4]. The orientation of the hydrides in the cladding can vary based on the stress states seen by the rod during service. The relationship between PCMI failure threshold and the extent of waterside corrosion and hydrogen pickup is clearly seen in both out of pile separate effects tests[5,6] and in the in-pile experimental database.[2] However, recent alloy developments by nuclear fuel vendors have resulted in zirconium alloys that are highly resistant to both waterside corrosion and hydrogen pickup[7]. Efforts are also currently underway to coat the cladding with a thin chrome layer which could all but eliminate corrosion concerns during normal operations.[8,9] PCMI failure thresholds derived from experiments on heavily corroded legacy claddings may be unnecessarily conservative when applied to these more modern alloys, which may allow PCMI failure threshold to be replaced by less limiting high temperature phase failure thresholds.

In a recent update to its "State of the Art Report on Fuel Performance in Reactivity Initiated Accidents" the NEA's Working Group on Fuel Safety (WGFS) identified representativity of experiments to prototypical RIA as a vital consideration.[10] The transient pulse width is an important factor to consider when comparing experiments to prototypical RIA conditions. It is theorized that longer pulses should decrease the likelihood of PCMI failure due to two factors. First the smaller strain rates should result in lower cladding stress by giving the material time to deform plastically. Additionally, longer transients should allow more time for heat transfer to occur from the pellet to the cladding, increasing the cladding's temperature and further promoting ductile behavior and decreasing the overall stress. The effect of pulse width can be predicted using modern fuel modeling and simulation codes[11] and has been confirmed in out of pile separate effects tests.[12] However, the pulse width affect cannot be deduced from the existing in-pile experimental database mainly due to the lack of experiments conducted between ~60ms and ~500ms as seen in Figure 1.[13] Modern 3-dimensional core physics calculations show increasingly wider pulse widths highlighting the need for experimental data in this area.

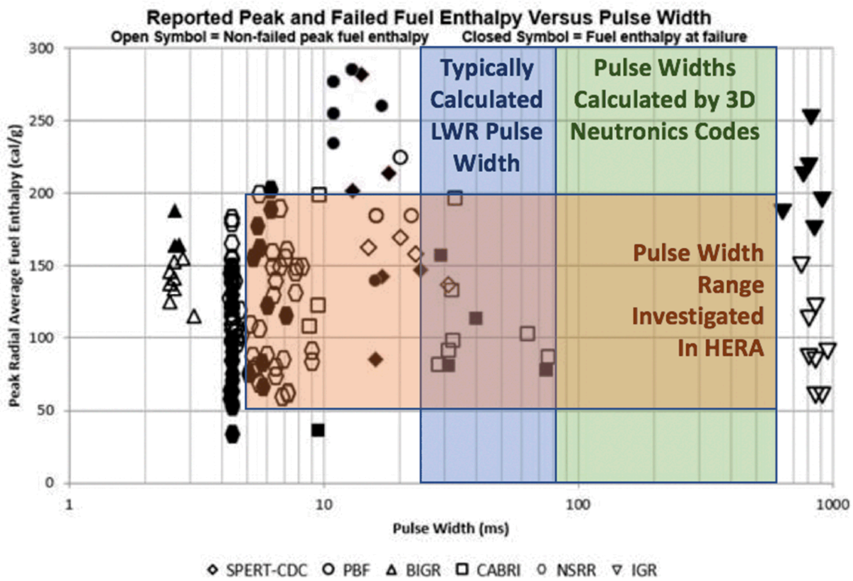

Figure 1. In-Pile RIA Testing Database as a function of Pulse Width.

Because modern high burnup fuels are expected to be more resilient to PCMI failure, the dominant failure mode may transition to ballooning and rupture failure. High burnup fuels contain an proportionally greater amount of fission gas stored in the fuel matrix. This fission gas can be suddenly released in the RIA transient due to high temperature exposure resulting in high plenum pressures that promote ballooning and rupture failure. Very little in-pile data exists to support the development of a swelling and rupture failure threshold for high burnup fuels.

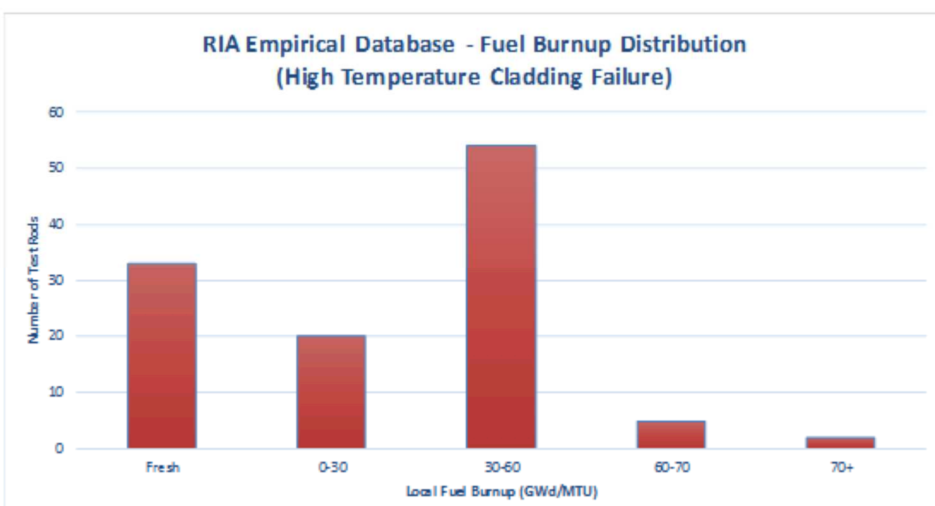

Figure 2. Burnup Distribution of Ballooning and Rupture Failures in RIA Experiments

Additionally, ballooning and rupture failure occurs with high hoop strains creating balloons similar to those often observed in loss of coolant accident (LOCA) transient simulations. In addition, it has been observed in prior LOCA simulations on high burnup fuels that the loss of cladding restraint promotes gross fuel fragmentation, relocation, and dispersal (FFRD).[14] In high energy RIA transients this could also lead to fuel melting in the ballooned region and upon quench, higher levels of fragmented and molten fuel coolant interaction than what has been observed in experimental programs to date. These interactions could pose a challenge to maintaining coolable geometry. Due to limitations of the experimental facilities previously available for testing, the current database of in-pile RIA experiments on high burnup fuels contains very few experiments with peak radial average enthalpies above ~100 calories per gram. Therefore, investigation of peak radial-average fuel enthalpy up to the current fuel failure limit of 150 cal/g-UO2 and to the core coolability limit for fresh and low burnup fuel of 230 cal/g-UO2 is needed to establish a high temperature failure limit for RIA events.

2. PROJECT OBJECTIVES

When considering the performance of modern high burnup fuels in RIA transients, application of PCMI failure thresholds may be overly conservative when applied to modern cladding alloys and RIA transients characterized by longer pulse widths. However, current core coolability limits have not been evaluated in integral experiments as applied to high burnup fuel that experiences ballooning and rupture and potential FFRD during transients with high energy releases. To address these concerns the HERA research program has two key research objectives:

1. Identify pulse width impacts or thresholds in in-pile experiments when considering the vulnerability of LWR fuel to failure by PCMI in RIA transients.
2. Provide relevant data to support maintaining core coolability associated with high burnup fuels that fail by swelling and rupture, and the impacts of potential resulting FFRD.

3. SCOPE OF WORK

The FIDES HERA program would ideally extend to multiple existing transient reactor facilities that would eventually form the foundation of FIDES' RIA 'core group' including the CABRI facility in France, the Nuclear Safety Research Reactor (NSRR) in Japan, the Annular Core Research Reactor (ACRR) in the USA, and the Transient REActor Test (TREAT) facility in the USA. In the initial three-year program of HERA, in-pile RIA irradiations will take place primarily at the TREAT reactor at Idaho National Laboratory (INL).

TREAT is an air-cooled reactor driven by a core of graphite blocks having a small concentration of dispersed uranium dioxide. Experiment assemblies are typically removed from or placed into the core through a slot in the reactor's upper rotating shield plug. Pulse type transients designed to simulate an RIA are initiated in TREAT by bringing the reactor to a low steady state power of 50 watts and then rapidly removing transient control rods, resulting in a step insertion of excess reactivity. TREAT pulses initially have a nominally Gaussian shape followed by a decaying exponential tail. Larger step reactivity insertions result in transient pulses that have higher peak powers, higher overall energy releases, and shorter pulse widths. TREAT has the ability to re-insert the transient control rods and shorten the natural transient. The clipping system has the ability to reduce both the pulse width (FWHM) and total energy released. As an example, for a 4.5% $\Delta k/k$ reactivity insertion, clipping capabilities could decrease the maximum energy released from the reactor from ~2800 MJ to ~630 MJ and shorten the pulse width from 103ms to 95ms. This approach gives the ability to tailor transients to a desired duration and energy release. Tailoring is currently limited by the speed of the rod drive system (~355 cm/s). For TREAT's ~1m reactor length, it takes ~280ms for the control rods to fully insert themselves. However, pulse widths of 89 ms have recently been demonstrated, and planned plant modifications involving a He-3 clipping system are predicted to be able to achieve pulse widths as low as 45 ms,[15] within reach representative PWR HZP RIA pulse widths and overlapping nicely to the upper end of pulse widths that CABRI, NSRR, and ACRR are capable of producing.

3.1 Experiment Description

Currently, experiments in TREAT are performed in a static water capsule called SERTTA (static environment rodlet transient testing apparatus) shown below in Figure 3. The SERTTA capsule is capable of elevated temperature and pressure testing creating an environment analogous to a pressurized or boiling water reactor. The capsule is equipped with an expansion chamber above the test specimen to allow for the rapid vaporization of the water during the transient. A high temperature crucible is then placed at the bottom of the capsule to safely accommodate partial or complete melting of the test specimen. The test rodlet has prototypic PWR radial dimensions (9.5mm outer diameter) and is 15cm long with a 10cm fueled length. SERTTA fits inside a containment pipe which is lowered into the reactor as an integral unit through the rotating shield plug. Twelve 1 mm instrument leads penetrate the capsule and containment and can be used to accommodate a variety of different instruments.

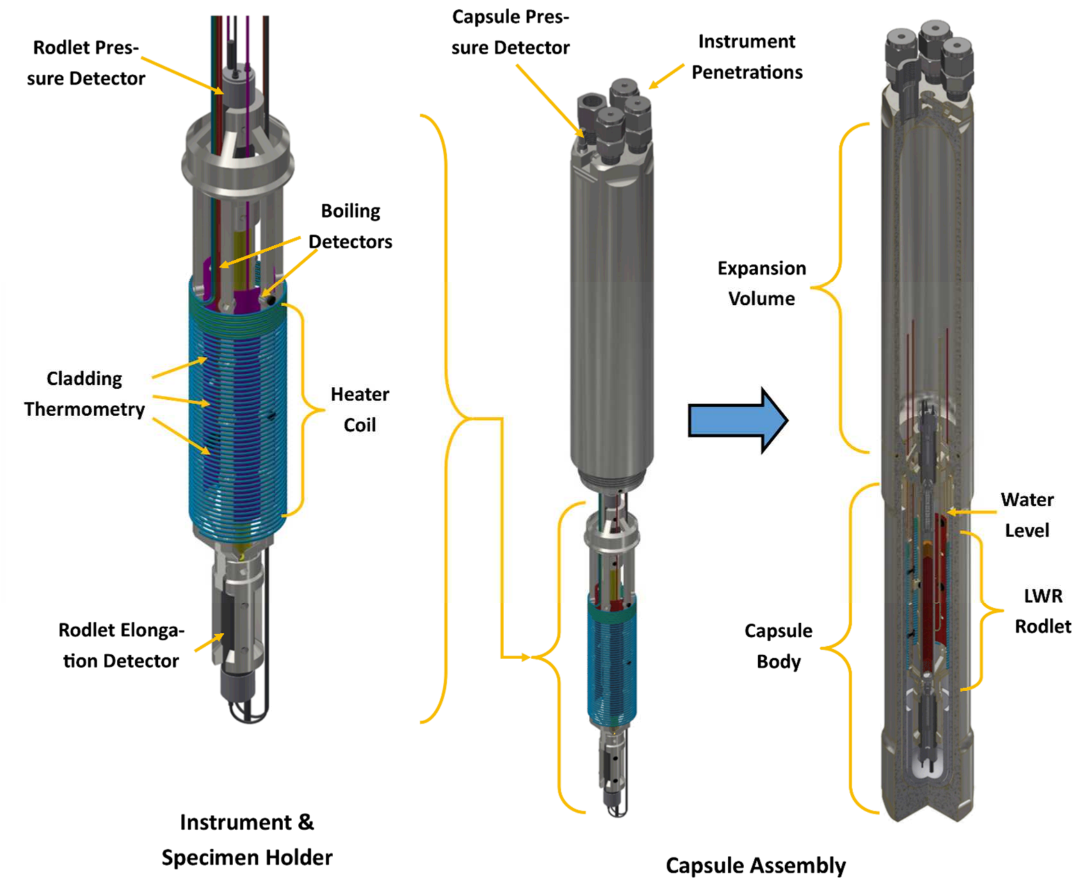

Figure 3. SERTTA Water Capsule.

To provide a clear analytical connection to the complexity of high burnup fuel behavior, carefully prepared experiments using well defined materials, geometries and boundary conditions will be used to support development of mechanistic understanding of RIA behaviors. Initial irradiations will involve fresh fuel with simulated high burnup characteristics. The cladding will be pre-oxidized and hydrided to simulate the effects of water side corrosion. The thickness of the oxide layer as well as the hydride concentration and morphology will be selected by the HERA core group advisory board and will utilize the latest methods for obtaining representative simulated environmental degradation in the cladding. The fuel rodlet will be fabricated with oversized UO2 (to simulate the reduced pellet-clad gap typical of high burnup fuel and subsequent closure of the gap) and will be over pressurized (to simulate high fission gas release). In this way the fresh fuel will be vulnerable to both PCMI failure as well as ballooning and rupture failure. Tests will be conducted with sufficient peak radial average enthalpy targets to fail the fuel rod. The tests will be conducted at different pulse widths to determine if a pulse width dependence or threshold on different failure modes can be determined. Tests down to approximately 50ms can be performed at the TREAT facility. Tests at lower pulse widths will be required to take place at different reactor. Possibilities include the Annular Core Research Reactor (ACRR) at Sandia National Laboratory in New Mexico in the United States, the CABRI reactor in France, or the Nuclear Safety Research Reactor (NSRR) in Japan. A proposed test matrix for the fresh fuel tests is presented in Table 1 below.

Table 1. HERA Fresh Fuel (Simulated High Burnup) Test Matrix.

Test Number	**Pulse Width (ms)**	**Reactor**
HERA-Sim-1	600	TREAT
HERA-Sim-2	300	TREAT
HERA-Sim-3	100	TREAT
HERA-Sim-4	50	TREAT
HERA-Sim-5	30	ACRR/CABRI/NSRR
HERA-Sim-6	5	ACRR/CABRI/NSRR

High burnup material irradiated in a commercial nuclear reactor will be tested to validate findings from the fresh fuel tests and investigate fuel failure core coolability thresholds at high burnup. The material identified for HERA is planned to be shipped to INL from the Byron Nuclear Generating station in Illinois and includes material irradiated above the current regulatory threshold of 62 GWD/MTU. Four tests are planned for high burnup material specimens. Two of the tests will be near radial average enthalpies that coincide with the current regulatory thresholds for fuel failure by swelling and rupture, and two will be at higher radial average enthalpies to address core coolability behaviors. The pulse widths for the transients will be selected by the HERA core group advisory board and will be based on the observed outcomes of the fresh fuel tests above as well as the existing state of knowledge. The proposed test matrix is presented in Table 2 below.

Table 2. HERA High Burnup Fuel Test Matrix.

Test Number	Burnup (GWD/MTU)	Target Peak Radial Average Enthalpy (cal/g)
HERA-HBU-1	60-70	170-200
HERA-HBU-2	60-70	200-230
HERA-HBU-3	70-80	170-200
HERA-HBU-4	70-80	200-230

Following the completion of the initial three-year phase of HERA future testing will extend to BWR fuel with Zircaloy-2 cladding, as well as accident tolerant fuel (ATF) materials irradiated at moderate and high burnup ATF materials of most immediate interest will be coated Zircaloy claddings with high density or doped UO_2 fuel pellets. Durring HERA phase 1, materials for these follow on studies will identified and the HERA core group will identify a rod harvesting strategy to receive these materials.

3.2 Experiment Tasks

The HERA project is broken down into three key R&D tasks in addition to a task devoted to management and administration of the project. The project work breakdown can be seen below in Figure 4. Task 1 involves the management, administration, and reporting for the project. Task 2 involves the execution of the fresh fuel experiments with four sub-activities: (1) conducting experiments at TREAT, (2) conducting experiments at a partner narrow pulse width facility, (3) conducting post transient examination of the experiments, and (4) conducting a round robin modelling and simulation benchmark of the fresh fuel tests. The benchmark modeling activity on the fresh fuel tests is ideal because experiment parameters will be well defined such as cladding hydrogen level and state of pellet cladding gap. This will address some critical gaps identified in efforts to validate modern modeling and simulation tools used to predict response of fuels to RIA events.

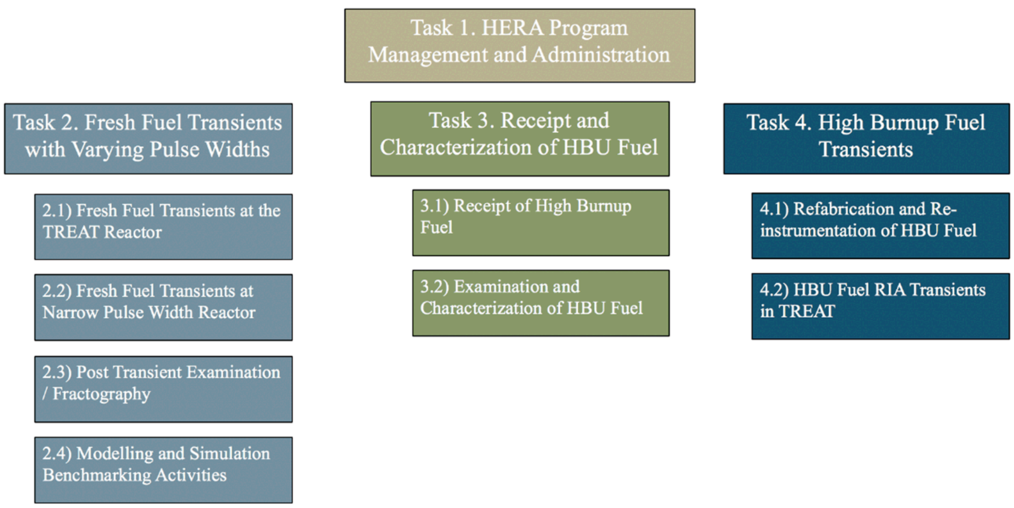

Figure 4. HERA Project Work Breakdown Structure.

Task 3 involves the receipt and characterization of the high burnup fuel that will be used in the high burnup experiments. Sub-activities include (1) receipt of the material from the Byron reactor, and (2) examinations of the parent fuel rods used in the test. Standard examinations of the parent rods will include visual exams, profilometry, axial gamma scanning, and eddy current oxide thickness. Following the non-destructive examinations, the rods will be punctured to determine plenum pressure, fission gas composition, and internal volume. Rods will then be sectioned and characterized using standard optical microscopy practices. Some mechanical testing of adjacent cladding samples will also be considered,

Task 4 involves the refabrication of the irradiated fuel rod segments and the execution of the high burnup transient irradiation experiments. The ~15cm segments will have the ends defueled resulting in an approximately 10cm fueled length. New endcaps will be welded onto the rod and instrumentation will be attached as necessary. The refabricated rodlets will be backfilled with a pressure equal to that of the parent rod with a gas mixture of helium and argon to represent the relevant behaviors of the original fission gas makeup. The rods will then be loaded into experimental hardware and sent to TREAT for transient irradiations. Online measurements will provide data on rod temperature and rod failure as well as the effects of any fuel dispersal and FCI on capsule pressure. Additionally, TREAT's fuel motion monitoring system will provide a temporal description of potential fuel motion/dispersal during the transient. Detailed, post transient examinations will likely need to be delayed until a subsequent phase of HERA.

4. SCHEDULE BUDGET AND DELIVERABLES

The entire project is expected to take three years to complete and will start in January of 2021 and conclude in December of 2023. The project budget is estimated to be 7.5 million USD. $3.75 million will be funded by the HERA core group and $3.75 million will be funded by FIDES, a 50/50 cost split. The majority of the HERA core group funding is anticipated to come from the U.S. Department of Energy (DOE). A project schedule and cost breakdown for each of the principal tasks is presented below in Figure 5.

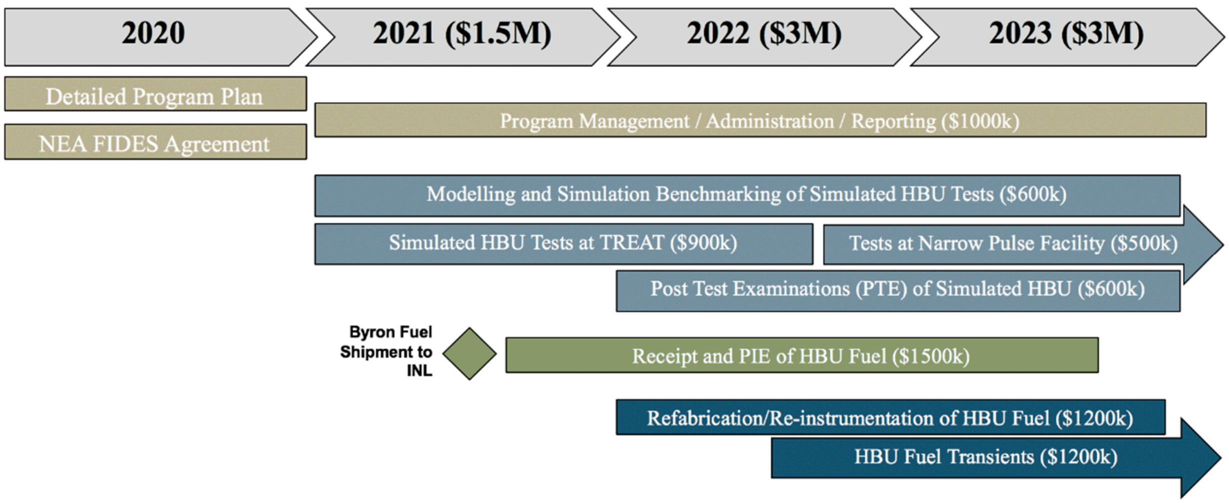

Figure 5. HERA Project Schedule and Budget Broken out by Task.

Deliverables for the HERA project will include 5 bi-annual progress reports beginning six months after the start of the project in June of 2021, with the last progress report being delivered six months prior to the conclusion of the of the of the project in June of 2023. The final project report will be summarizing the project results and conclusions will be delivered in draft form at the conclusion of the project in December of 2023. The final report will include detailed appendixes for each of the three principal tasks that are also capable of functioning as stand-alone reports. After incorporating comments from a three-month review period, the final HERA report will be issued by the end of March 2024. All fixed and raw data generated during the HERA project will be subjected to internal review at INL following INL's conduct of research principals and quality assurance program. Data will then be stored in INL's Nuclear Data Management Systems (NDMAS) database which is an NQA-1 data storage system. From there data can be transferred to the FIDES database that is being developed by NEA.

5. PROJECT GOVERNANCE, AND RELATIONSHIP TO OTHER PROJECTS

The HERA project will be executed almost entirely at INL and therefore INL will operate as the central operating agent for the project. INL is the United States' lead laboratory for Nuclear Energy research and development. INL is an engineering laboratory with multiple test reactors and hot cells to conduct nuclear energy engineering research projects. INL is located in the state of Idaho in the mountain west region of the United States near the town of Idaho Falls.

Overall responsibility for HERA project execution will reside with the HERA Principal Investigator (PI) who will be supported by an Experiment Manager (EM), both of whom will be INL staff. The HERA PI will report to a management board made up of a total of 6-10 individuals representing all of the core group member organizations. The HERA core group will be initially comprised of the U.S. Department of Energy (DOE), the U.S. Nuclear Regulatory Commission (NRC), the Electric Power Research Institute (EPRI), the French Institut de Radioprotection et de Surete Nucleaire (IRSN), and the Japanese Atomic Energy Agency (JAEA). The HERA core group management board will be responsible for overseeing the project to ensure that the overall objectives are being achieved and the schedule and budget are being maintained. The management board will nominate an individual to serve as the chair of the management board. The chair of the management board and the HERA PI will be responsible for communicating project progress to the FIDES governing board, which will be responsible for funding roughly 50% of the project. The HERA project is closely related to and synergistic with several other national and international research endeavors, namely (1) the CABRI International Project (CIP) executed by IRSN in France, (2) the Advanced LWR Fuel Performance and Safety (ALPS) program executed by JAEA in Japan, and (3) the Accident Tolerant and High Burnup Fuels program sponsored by the U.S. DOE. The HERA core group management board will be responsible for ensuring that the project objectives of HERA remain relevant in light of the R&D findings and direction coming from these other connected research projects and programs. Figure 6 below shows the overall HERA governance structure including the relationship to these other projects and programs.

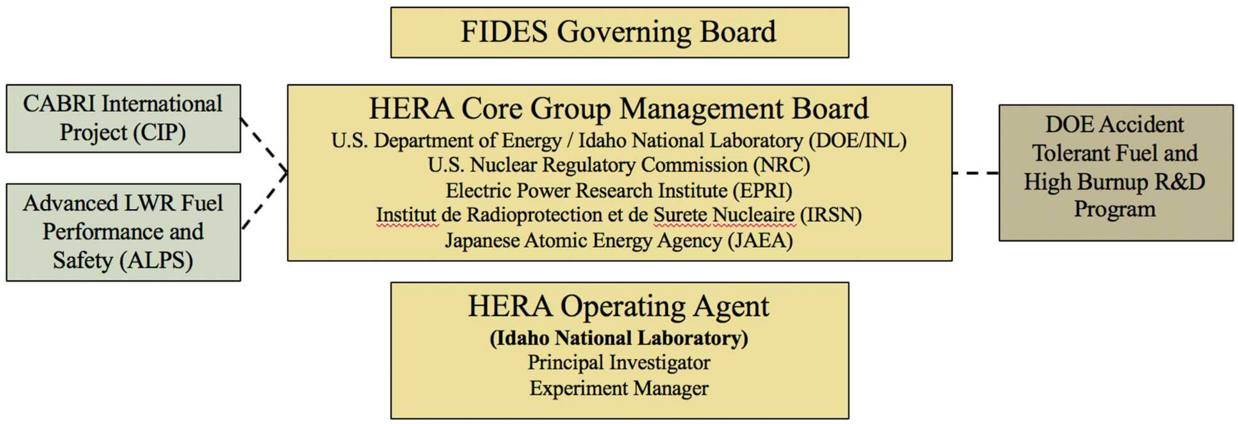

Figure 6. HERA Governance and Management Structure.

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