STATEMENT OF OBJECTIVES

The U.S. Department of Energy entered a cooperative agreement with ASME Standards Technology, LLC (ASME ST-LLC) to update and expand appropriate materials, construction and design codes for application in future Generation IV nuclear reactor systems that operate at elevated temperatures. The scope of work, including performance measures, milestones, and deliverables, is divided into specific tasks that are described below. These tasks are tied to the Generation IV Reactors Integrated Materials Technology Program Plan. The tasks with the highest priority are listed first. The highest priority tasks should be given preferential treatment whenever funding or resource constraints arise.

Tasks developed and approved for funding (subject to availability) in subsequent fiscal years will be amended to this statement of objectives to establish additional measures, milestones, and deliverables that are to be accomplished under this cooperative agreement. Some of these tasks, for example Task 1 and Task 10, are expected to be multi-year tasks. Some lower priority tasks may not be funded due to the limitation of funding and the higher priority of other tasks.

TASK A: Overall Management and Operation of Generation IV Material Tasks

This task includes all the activities that are needed to develop specific scope, cost, and schedule proposals for these and other, future Generation IV materials-related tasks. It includes the ASME management of all of the tasks under this agreement, such as status reporting, financial reporting, and project management and control.

TASK B: Participation in Kickoff Meeting and Semi-annual Review Meeting

ASME's project management over these tasks shall participate in an initial project meeting, and at project review meetings every six months as required by DOE. The meetings will be at a location determined by DOE but could be in Washington, D.C., Idaho Falls, Idaho, or Oak Ridge, Tennessee. The purpose of these meetings is to provide DOE the opportunity to review the progress of the tasks, remain involved in the direction of the activities in a substantial way, and ensure that outcomes address and support the Generation IV Reactors Integrated Materials Technology Program Plan

TASK 1: Verification of Allowable Stresses in ASME Section III, Subsection NH With Emphasis on Alloy 800H and Grade 91 Steel (a.k.a., 9Cr-1Mo-V or "Modified 9CR-1Mo")

<u>Part I: Review of alloy 800H stress allowables in ASME III-NH and extension of time-dependent allowable stresses to 900°C</u>.

Secure formal access and use rights of the various materials databases. Assemble the original database and review the methods used to set the time-dependent allowables. Review the European and Japanese databases and compare the methods and procedures used by these sources with the methods used to set stress for III-NH. Augment the US database to include data produced to 900°C. This database will include both creep and stress rupture data. Assess alternate procedures for describing creep and stress-rupture for conditions of concern to the VHTR. Procedures developed by the Pressure Vessel Research Council will be included in this assessment. Compare the current allowables to the results of the re-assessment and recommend a course of action with respect to II-D and III-NH. This course of action may identify supplementary testing to address the needs outlined in the High Temperature Metallic Materials Test Plan for Generation IV Nuclear Reactors.

Part II. Review of the strength of weldments for alloy 800H and extension of technology to 900°C.

Secure formal access and use rights of the various materials databases. Assemble the original database and review the methods used to set weldment strength factors. Review the European and Japanese databases and compare the methods and procedures used by these sources with the methods used for III-NH. Augment the US database to include data produced to 900°C. This database will include both creep and stress rupture data. Recommend a course of action with respect to III-D. This course of action may identify supplementary testing to address the needs outlined in the High Temperature Metallic Materials Test Plan for Generation IV Nuclear Reactors.

Part III. Review of the allowable stresses for Grade 91 steel

Secure formal access and use rights of the various materials databases. Assemble an up-dated compilation of the creep and rupture data. This compilation will be especially important for thick plate, forgings, and heavy wall piping. Review the European and Japanese databases and compare the methods and procedures used by these sources with the methods used to set stress for III-NH. Assess alternate procedures for describing creep and stress-rupture life for conditions of concern to the VHTR. Procedures developed by the Pressure Vessel Research Council will be included in this assessment. Compare the current allowables to the results of the re-assessment and recommend a course of action with respect to II-D and III-NH. This course of action may identify supplementary testing to the address the needs outlined in the High Temperature Metallic Materials Test Plan for Generation IV Nuclear Reactors.

Task 1 Deliverables:

Part I: Review of alloy 800H stress allowables in ASME III-NH and extension of time-dependent allowable stresses to 900°C.

1. A report will be prepared that compares databases, analysis methods, and allowables for US, European, and Japanese codes.
2. A report will be prepared that includes an extended database to 900°C and assesses procedures for setting allowable stresses to 900°C.
3. Since the proposed task schedule spans more than one fiscal year, an interim report deliverable will be provided to document first-year progress.

Part II. Review of the strength of weldments for alloy 800H and extension of technology to 900°C.

1. A report will be prepared that reviews the database for alloy 800 weldments that was used to assess weld issues for III-NH and compares III-NH with European and Japanese codes.
2. A report will be prepared that recommends testing to extend the data base to 900°C and assesses procedures for setting allowable stresses to 900°C.

Part III. Review of the allowable stresses for Grade 91 steel

1. A report will be prepared that reviews the database, summarizes the technology leading to the current allowables, and defines issues to be resolved regarding these stress allowables.
2. A report will be prepared that assesses analysis methods for estimating creep and stress-rupture life for conditions of interest to the VHTR. (This report will serve as a basis for the recommendation of Code action on allowables.)
3. In consort with other tasks, recommendations will be developed for uniaxial testing to gather the type of baseline data needed to address Probability Risk Assessment and NRC issues.

TASK 2: Regulatory Safety Issues in Structural Design Criteria of ASME Section III Subsection NH and for Very High Temperatures for VHTR & GEN IV

Identify the safety issues relevant to this section of the code that must be resolved for licensing purposes. This task includes developing a description of how Subsection NH of the ASME Code addresses these issues and the further needs to add criteria to cover unresolved safety concerns for Very High Temperature Service.

The major fundamental regulatory safety need is for improving the criteria to prevent creep cracking. An essential element is the need to build confidence in the regulatory community that the resulting designs will have adequate safety margins. In improving and developing these new criteria and stress/strain limits, the NRC must be consulted. To this end, initiate current communications with the NRC on this issue and make an evaluation of the historical record.

Task 2 Deliverables:

1. A description of the high temperature structural integrity safety concerns raised by U.S. NRC and ACRS senior technical people.
2. A report describing how issues described in Deliverable 1 are addressed in subsection NH. The report should address any new criteria that are needed to prevent cracking in very high temperature service.

TASK 3: Improvement of ASME Subsection NH Rules for Grade 91 Steel—(negligible creep and creep-fatigue)

Part I: Negligible creep regime

Examine current approaches available to define negligible creep and check their applicability to Grade 91 steel. The work will be based on material data available in France and the U.S. The output of this work will be a program identifying the tests required to support the definition of negligible creep conditions for Grade 91 steel.

Part II: Creep-fatigue procedure

Compare Subsection NH and RCC-MR creep-fatigue procedures. Comparisons will be performed on cases defined on the basis of experimental test results available from Japan, France and the U.S. on Grade 91 steel. Particular attention will be paid to the definition of safety factors and creep-fatigue damage envelope. Explore the extent to which material data presently available in nuclear codes are thought to be validated. Recommend improvements to existing procedures and define a test program to validate the improved procedures.

Task 3 Deliverables:

Part I: Negligible creep regime:
Provide a synthesis report on negligible creep and provide a recommendation for a test program to support the definition of negligible creep conditions.

Part II: Creep-fatigue procedure:
Provide a report that compares existing procedures and applicability of creep-fatigue to Grade 91 steel, and provide a recommendation for a test program to validate existing or suggested/new improved creep-fatigue procedures.

TASK 4: Updating of ASME Nuclear Code Case N-201 to accommodate the needs of High Temperature Gas Cooled Reactors currently in development

The scope of the code case needs to be expanded to include the materials with higher allowable temperatures or extend the temperature limits of current materials and to confirm that the design methodology used is acceptable for design of core support structure components at the appropriate elevated temperatures.

1. Identify the maximum operating temperature required for HTGR metallic core support structures (review data from Areva, GA, PBMR, DOE etc.) Define operating parameters, temperature, pressure, environment etc.

2. Identify candidate materials for use as metallic core support structures within the defined operating parameters (Grade 91 steel, alloy 617, Hastalloy X/XR etc.)

3. Confirm design methodology to be used. Compare with any proposed changes to Subsection NH.

4. Define material data required.

5. Search published data for adequacy; carry out gap analysis.

6. Define material testing required to obtain data required.

Task 4 Deliverables:

A final report documenting the findings of the work under this task, addressing each of the six areas above.

TASK 5: Collect Available Creep-Fatigue Data and Study Existing Creep-Fatigue Evaluation Procedures for Grade 91 Steel and Hastelloy XR

Secure formal access and use rights of the various materials databases. Evaluate the state of existing data and determine where more data is necessary. The necessary creep-fatigue data on Grade 91 steel and Hastelloy XR will be collected. Grade 91 steel was a candidate material for Japanese demonstration fast breeder reactor and Hastelloy XR is used in the High Temperature Gas Reactor (HTTR). Data to be collected involve number of cycles to failure under various conditions. Data on cyclic inelastic behaviors will also be collected when available. It remains to be determined if Hastelloy XR data can be transferred directly to the United States for use by ASME. The transfer of data is to be determined and negotiated.

Creep-fatigue criteria will be summarized based on existing codes and standards. Creep-fatigue damage evaluation procedures will be compared to each other. Points to be addressed will include assessment methods of strain range, initial stress and relaxation behavior, along with the formulation of creep damage.

Based on the information collected, an outline of research and development items that are necessary in relation to creep-fatigue evaluation of Grade 91 steel and Hastelloy XR under the conditions expected in GEN IV and VHTR reactors will be described. The effort on Grade 91 steel will be closely integrated with a related task 3 (above) on Insignificant Creep and C-F of Grade 91 steel, particularly the C-F evaluation of U.S. and French data on Grade 91 steel.

Task 5 Deliverables:

A report that addresses the following:

1. The status of Japanese creep-fatigue data on Grade 91 steel and Hastelloy XR.
2. The status of existing creep-fatigue criteria related to Grade 91 steel and Hastelloy XR.
3. An outline of research and development items in relation to creep-fatigue evaluation of Grade 91 steel and Hastelloy XR under the conditions expected in GEN IV and VHTR reactors, and possible applicability to Haynes 230 and Inconel 617.

TASK 6: Graphite and Ceramic Code Development

Several new graphite and ceramic materials are being identified in new nuclear power plant designs because of high temperature operating environments. Requirements must be established, and environmental effects must be considered, including irradiation to support emerging designs and maintain the technical edge for standards development in the international community.

Through these sub-tasks, this task will, support development of Code requirements for qualification of non-metallic components, including carbon-carbon composites and ceramics such as SiCSiC composites for new nuclear power plant designs.

The project team [Section III – Project Team on Graphite Core Components (N20070348)] is currently drafting design rules for the graphite core internal for a VHTR. It is the project team's intent to eventually develop design rules for the composite and ceramic core internals.

Several subtasks are identified to support the work of the project team:

1. Provide independent peer review of the draft code rules, including a framework of probabilistic design approaches, fracture mechanics assessment methodologies, qualification of components, material specification/traceability, examination or inspection methods and acceptability criteria, and irradiation considerations in design rules.
2. Provide independent consulting services (including NASA Glen Research Center, and other independent consultants as identified by Section III) to the Section III committee.
3. Organize workshops to inform ASME staff and committee members of the technical and background information needed regarding the behavior and properties of brittle materials such as graphite and ceramics, including the effects of neutron damage on their properties.
4. Review adequacy of current data base and make recommendations regarding further data needed to support the design rules
5. Review and endorse test plans and programs, both within the U.S. and internationally, that provide supporting data for the code.

Task 6 Deliverables:

1. Organize and hold a minimum of 2 workshops as indicated above, in subtask #3 and provide a report summarizing major outcomes of discussions and presentations; report should include a copy of all presentation materials.
2. A report from the peer review team summarizing the evaluation and recommendations of draft code rules as outlined above in sub tasks #1, #4, and #5.
3. A report summarizing the reviews and the recommendations resulting from sub task #2

TASK 7: NH Evaluation and Simplified Methods

Part I: Simplified and Inelastic Design Methods

Review and compare current design methods in ASME NH, RCC-MR, BC5500, DIN and JNC. Assessment techniques in API 579 and R5 are also relevant.

Propose a range of design analysis methods consisting of:

Elastic analysis.
Limit, shakedown and ratcheting analysis.
Full inelastic analysis.

Part II: Creep-Fatigue Design and Assessment

Review creep-fatigue methodologies, including crack growth, damage-based and strain-based methods. Likely sources will include the above-mentioned sources, GE Report DOE-ET-34202-80 and ORNL-5073. Identify applications and areas of difficulty in connection with Grade 91 steel and Alloy 617/230/800H materials. Include aging, crack initiation, surface and environmental effects on these materials. Critically evaluate data and methodology in the light of likely VHTR cycles and assessment requirements. The report will comment on the adequacy of existing methods and will include recommendations to address problems. These could include life prediction models, extrapolation of data, test data and techniques. This task will not be conducted in vacuum relative to other tasks that address creep-fatigue, rather it serves as a parallel but non-duplicate path at addressing creep-fatigue. Addressing such a complicated problem with several different concepts is desired.

Task 7 Deliverables:

Part I: Design Methods:
A final report with recommendation on design methods and requirements for supporting data.

Part II: Creep-Fatigue Design and Assessment
A final report with recommendations on testing approaches and needs, and life prediction models.

TASK 8: Identify Future Test Needs to Validate Elevated Temperature Design of VHTR

This task identifies the future test needs to validate elevated temperature design procedures.

1. Review VHTR (block and pebble designs) design features from the viewpoint of elevated temperature design, and summarize points to be validated focusing on major components such as reactor vessel, internal structures, piping, etc.
2. Review what has been accomplished in the past to validate existing design procedures and related activities for which results would contribute to the validation.
3. Based on the information obtained by the above review, describe future test needs for the validation. Provide very rough cost estimate and time schedule as well.

Task 8 Deliverables:
A report that summarizes the findings from the review above, including identification of future test needs for validation and rough cost estimate and time schedule.

TASK 9: Environmental and Neutron Fluence Effects in Structural Design Criteria of ASME Section III Subsection NH and for Very High Temperature VHTR & Gen IV Designs

Part I: Evaluate the needs for inclusion of environmental and neutron fluence effects in design criteria for VHTR core support structures

Evaluate the role of Environmental Degradation and Neutron Fluence Degradation in the design for metallic core internal structures of high-temperature gas-cooled reactors. These effects have not been incorporated into construction codes (ASME, German, Japanese, etc.). The approach taken for liquid metal fast breeder reactor components may provide a basis for supplementing the rules that are now in III-NH.

Obtain VHTR conceptual design information essential to the development of supplemental design criteria. This information will include materials of construction, the expected range of temperatures, expected range of gas compositions, expected neutron fluence, capabilities for component replacement, and the like.

Review the plans of US Gen-IV materials program to access the adequacy of these plans to develop the materials performance characteristics needed to formulate supplemental rules to address environmental and neutron effects. The materials of emphasis will be UNS N06617, N06230, N06002, N08810, and alloy XR.

Part II: Review of stress design criteria for VHTR reactor pressure vessel design considering environmental and neutron fluence effects.

Review the current U.S. European and Japanese materials selection criteria and conceptual designs of reactor pressure vessels for gas-cooled reactor applications. This review will include applicable Design Codes.

Review the Generation IV Reactors Integrated Materials Technology Program Plan to assess the adequacy of the plan to produce the information needed for necessary supplementary rules for alternate VHTR concepts. Pressure vessel steels will include SA508/SA533 and Grade 91 steel.

Task 9 Deliverables:

Part I: Evaluate the needs for inclusion of environmental and neutron fluence effects in design criteria for VHTR core support structures.

A Report describing the needs and outlining the content of a supplementary design criteria document that includes environmental and neutron effects considerations for VHTR core support structures.

Part II: Review of stress design criteria for VHTR reactor pressure vessel design considering environmental and neutron fluence effects.

A Report that provides current best practice supplementary design criteria for environmental and neutron effects considerations for VHTR reactor pressure vessels.

TASK 10: Development of ASME Code Rules for the Gas Cooled Reactor Intermediate Heat Exchanger (IHX)

The intent of this task is to determine how and where within ASME codes and standards the IHX, safety valve, etc. would be addressed. In order to answer this question, many technical questions need to be addressed to determine how the function of such components affects the plants, safety, etc.

<u>Part I. Review of Current Experience.</u>

The objective is to identify all aspects: materials, design, fabrication, examination, testing, overpressure protection and inservice inspection; used in the construction and operation of representative heat exchanger pressure boundary and internals designs. Heat exchangers with working fluid temperatures at the upper end of the creep regime for their materials of construction will be emphasized. Candidate designs will include as a minimum the Heatric concept and a representative plate/fin concept. Additional concepts will be identified if possible. Particular attention will be paid to (a) design criteria including methods, if any, for evaluation of cyclic life, (b) construction codes of record and designated pressure boundaries and (c) qualification of materials and fabrication techniques for the intended service.

<u>Part II. Assessment of Current IHX Concept(s)</u>

This will start with an evaluation of the current IHX design approach(s) with respect to the evaluation of past experience accomplished in the first task. This will include scoping analyses to identify critical design configurations and loading conditions. This will require close coordination with the reactor system designers and candidate component suppliers to ensure that the conditions evaluated are representative. Operational and safety aspects of a gas cooled reactor design will be evaluated in light of past practice and experience.

<u>Part III. Development of Recommended Code Approach</u>

Based on the above results, recommend changes and additions to current construction codes or features of a new construction code will be identified. As noted above, all aspects: materials, design, fabrication, examination, testing, overpressure protection and inservice inspection will be considered. Candidate Codes that will be considered include Subsection NB and NH, Subsection NC and ND and their respective elevated temperature Code Cases and Section VIII, Div 1 and 2. A rough order of magnitude estimate will be prepared for the cost of implementing the proposed approach.

Task 10 Deliverables:

<u>Part I. Review of Current Experience.</u>
Report on "Review of Current Experience"

Part II. Assessment of Current IHX Concept(s)
Report on "Assessment of Current IHX Concept(s)"

Part III. Development of Recommended Code Approach
Report on "Recommended Code Approach"

TASK 11: Flaw Assessment and Leak Before Break (LBB) Approaches in ASME

Provide additional definition for this task, specifically related to the definition of LBB. Currently LBB is not covered by the code.

The objective of the present proposal is to perform a status report of rules presently available and to propose recommendations for further work within ASME.

The work consists of making a synthesis of approaches available for LBB assessment and more generally for fracture mechanics methods (crack growth and stability calculations). The work should clarify to which extent existing methods would be applicable for HTRs applications.

The work should also identify material properties available. Materials of interest would include but not necessarily limited to SA 508 / SA 533 and Grade 91 steel.

The output of the work will be recommendations for the definition of rules to be introduced in the ASME Code. A program will be defined indicating necessary tests to be carried out to establish a set of material properties for flaw assessment methods and/or specific tests to validate LBB approaches for HTRs.

This work could be used as a basis for discussion with USNRC before launching significant activities on this subject.

Task 11 Deliverables:
A report on LBB and flaw assessment methods and recommendation of a test program to define material properties for flaw assessment methods and to validate LBB approaches.

TASK 12: Improved NDE Methods for Metals

This task could be revised to address the metallic NDE in conjunction with the metallic work being discussed in this project.

Improved NDE Methods for Metals

> Part I. Identify appropriate new construction and in-service NDE methods for examination of metallic materials (e.g. acoustic emission, ultrasonic). Sub tasks are as follows:

a) Define maximum acceptable flaw types and sizes based on the LRFD approach that is developed and the material properties of candidate materials have been obtained.

b) Define nondestructive examination methods needed to detect sub critical flaws of the size and type defined in a) above in pressure equipment during initial construction and for periodic examination during the life of the equipment. It is anticipated that new methods will be needed to reliably detect smaller discontinuities than those of concern for the current generation of pressure equipment. The methods will include the characterization of uncertainties in a manner that is suitable for reliability-based LRFD development. Some methods to be considered include:

   i) Ultrasonic Time-of-Flight-Diffraction – provide detailed guidance for application.

   ii) Ultrasonic Phased Arrays – define requirements.

Part II. Provide technical information and background to resolve concerns and assist codes & standards committees and jurisdictional authorities in adopting improved NDE methods into codes and standards.

Task 12 Deliverables:

Part I – Report on NDE methods for new construction and in-service inspection for use in providing technical information and background to resolve concerns and assist codes & standards committees and jurisdictional authorities in adopting improved NDE methods into codes and standards.

Part II – Report with additional technical information and background.