INEEL/EXT-02-00107

Feasibility Study of Supercritical Light Water Cooled Fast Reactors for Actinide Burning and Electric Power Production

Nuclear Energy Research Initiative Project 2001-001
Progress Report for Year 1, Quarter 1 (September through December 2001)

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Idaho National Engineering and Environmental Laboratory
Bechtel BWXT Idaho, LLC

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Project Description

The use of supercritical temperature and pressure light water as the coolant in a direct-cycle nuclear reactor offers potential for considerable plant simplification and consequent capital and O&M cost reduction compared with current light water reactor (LWR) designs. Also, given the thermodynamic conditions of the coolant at the core outlet (i.e. temperature and pressure beyond the water critical point), very high thermal efficiencies of the power conversion cycle are possible (i.e. up to 46%). Because no change of phase occurs in the core, the need for steam separators and dryers as well as for BWR-type re-circulation pumps is eliminated, which, for a given reactor power, results in a substantially shorter reactor vessel than the current BWRs. Furthermore, in a direct cycle the steam generators are not needed.

If a tight fuel rod lattice is adopted, it is possible to significantly reduce the neutron moderation and attain fast neutron energy spectrum conditions. In this project a supercritical water reactor concept with a simple, blanket-free, pancake-shaped core will be developed. This type of core can make use of either fertile or fertile-free fuel and retain the hard spectrum to effectively burn plutonium and minor actinides from LWR spent fuel while efficiently generating electricity.

This reactor concept presents several technical challenges. The most important are listed below.

1) **Fuel and Reactor Core Designs:**
   - Local or total coolant voiding increases leakage, but hardens the neutron energy spectrum and decreases parasitic absorption. The net effect can be a reactivity increase. The core must be designed to ensure that the overall reactivity coefficient is negative.
   - A low conversion ratio fuel rapidly loses reactivity with burnup, thus requiring a large excess reactivity at beginning-of-life to operate continuously for an acceptably long time. Therefore, a control system must be designed that safely compensates for reactivity changes throughout the irradiation cycle, or the spectrum must be hardened to increase the conversion ratio.
   - Because of the hard spectrum, the Doppler feedback will be much smaller than that found in typical LWRs.

2) **Fuel Cladding and Structural Material Corrosion and Stress Corrosion Cracking:**
   - Because of the oxidizing nature of high temperature water, corrosion and stress corrosion cracking of the fuel cladding and core internals materials are expected to be major concerns for this reactor concept.
   - Because of the hard neutron spectrum, radiolysis of the water coolant may take place at a higher rate than in traditional LWRs. In addition, the radicals formed by the radiolytic decomposition of the water are highly soluble in supercritical water and may not recombine as well as in an LWR.
   - The hard neutron spectrum makes the irradiation damage of the fuel cladding and core structural materials more pronounced than in traditional LWRs. Also, high-energy neutrons work as catalysts for the oxidation and stress corrosion cracking of the structural materials (irradiation assisted stress corrosion cracking).

3) **Plant Engineering and Reactor Safety Analysis:**
   - Depending on its mission (e.g. electricity generation, co-generation of steam and electricity, desalinization), the plant will exhibit different optimal configurations and operating conditions.
   - Because no change of phase occurs in the reactor vessel, the need for a pressurizer to maintain the operating pressure has to be assessed.
   - The implications of utilizing supercritical water on the design of the reactor containment need to be evaluated.

- Because of the significant coolant density variation along the core, the supercritical water reactor might be susceptible to coupled neutronic/thermal-hydraulic instabilities.
- The response of the plant to design and anticipated accidents and transients might differ significantly from that of LWRs and needs to be evaluated.

The project is organized in three tasks, reflecting the three technical challenges above.

Task 1. Fuel-cycle Neutronic Analysis and Reactor Core Design (INEEL). Metallic, oxide, and nitride fertile fuels will be investigated to evaluate the void and Doppler reactivity coefficients, actinide burn rate, and reactivity swing throughout the irradiation cycle. Although metallic alloy fuels are incompatible with the water coolant, we envision the use of a dispersion type of metallic fuel, which will be compatible with water. Included in the fertile options will be the use of thorium. The main variables are the core geometry (e.g. fuel rod length, pitch-to-diameter ratio, assembly configuration) and the fuel composition. The MCNP code will be utilized for instantaneous reactivity calculations and the MOCUP code for burnup calculations and isotopic content.

Task 2. Fuel Cladding and Structural Material Corrosion and Stress Corrosion Cracking (University of Michigan and MIT). The existing data base on the corrosion and stress-corrosion cracking of austenitic stainless steel and nickel-based alloys in supercritical water is very sparse. Therefore, the focus of this work will be corrosion and stress corrosion cracking testing of candidate fuel cladding and structural materials. In Year 1 of the project MIT will use an existing supercritical-water loop to conduct initial corrosion experiments on a first set of candidate alloys in flowing supercritical water, and will identify promising candidate alloys classes for core internal components and fuel cladding based on existing data on the alloys radiation stability and resistance to both corrosion and stress-corrosion cracking. A high temperature autoclave containing a constant rate mechanical test device will be built in Year 1 and operated in Years 2 and 3 at the University of Michigan. The resulting data will be used to identify promising materials and develop appropriate corrosion and stress corrosion cracking correlations.

Task 3. Plant Engineering and Reactor Safety Analysis (Westinghouse and INEEL). The optimal configuration of the power conversion cycle will be identified as a function of the plant mission (e.g. pure electricity generator, co-generation plant, hydrogen generator). Particular emphasis will be given to the applicability of current supercritical fossil-fired plant technology and experience to a direct-cycle nuclear system. A steady-state sub-channel analysis of the reactor core will be undertaken with the goal of establishing power limits and safety margins under normal operating conditions. Also, the reactor susceptibility to coupled neutronic/thermal-hydraulic oscillations will be evaluated. The response of the plant to accident situations and anticipated transients without scram will be assessed. In particular the following transients and accidents will be analyzed: start-up, shut-down, load change and load rejection; LOCAs and LOFAs. As part of this analysis, a suitable containment design will be explored to mitigate the consequences of LOCA accidents.

Accomplishments in Year 1, Quarter 1 (Sep01-Dec01)

Although officially initiated in September 2001, the project could effectively start at the University of Michigan, MIT and INEEL only at the beginning of November 2001 because of various administrative problems and the late inclusion of MIT in the research team. The project has not started at Westinghouse yet. Therefore, this report summarizes only the accomplishments made at the INEEL, University of Michigan and MIT in the months of November and December 2001.

Task 1. Fuel-Cycle Neutronic Analysis and Reactor Core Design (INEEL)

1.1. Fuel Parameters and Analysis Tools

A qualitative analysis has been performed to determine which fuel form would support the highest reactivity-limited burnup, and would have the most proliferation resistant isotopics at a particular burnup. This study will define the trends of the specific fuels that will be used in the full core analysis. For this study, a full-length pin cell analysis was used to accommodate the change in coolant density along the pin. The parameters of the pin cell and the beginning-of-life plutonium/minor actinide isotopic fractions can be found in Table 1.

Table 1. Fuel parameters used in the analysis.

Parameter	Value
Fuel Radius (cm)	0.368
Gap Thickness (cm)	0.02
Clad Outer Radius (cm)	0.44
Active Fuel Length (cm)	120
Pin Pitch - triangular (cm)	1.01
Fuel Temperature - average (K)	900
Number of Coolant Nodes	22
Pu and MA BOL Fractions	
Pu-238	1.6%
Pu-239	46.3%
Pu-240	20.8%
Pu-241	8.0%
Pu-242	3.2%
Np-237	8.5%
Am-241	9.1%
Am-242	0.0%
Am-243	1.8%
Cm-244	0.6%

The plutonium and minor actinides comprise 13-20wt% of the fuel, with the remainder of the fuel consisting of uranium or thorium as either mono-nitrides or in a zirconium-metal matrix. More specifically, the thorium-based fuels have a constant plutonium and minor actinide content of 20%, while the uranium-based fuel plutonium and minor actinide content varies.

The MOCUP (Moore et al. 1995) code was used to analyze the reactivity-limited burnup and isotopic content of the fuel. MOCUP employs MCNP (Briesmeister 1997), a well-known Monte Carlo code capable of calculating fluxes, reaction rates, and eigenvalues in general, 3-D geometry using continuous cross-section data; and the ORIGEN (Croff 1980) matrix exponential method code that calculates the

generation and depletion of isotopes, or elements, in a given neutron flux. MOCUP takes specific output data (including cross-section data, fluxes, and reaction rates) from MCNP and passes it to ORIGEN, where new isotopic information is generated and passed back to MCNP for the next calculation. This gives time dependent information about the reactivity swing and isotopics for the specified problem.

1.2. Reactivity Swing Analysis

Long core life is highly desirable from the standpoint of proliferation resistance and resource utilization. However, the practical material issues and safety issues associated with high neutron fluences and fuel behavior under extreme conditions may limit the core life. Nonetheless, it is important to perform scoping studies that can produce results that lie beyond current technologies. The work presented here assumes some fuel lifetimes in excess of 10 effective-full-power-years, although no calculations were performed for times longer than 10 years.

Figures 1 and 2 show the reactivity versus the effective-full-power-years (EFPY) for both nitride and metallic fuel, respectively. Note that the pin was held at a constant power of 24 kW, or a linear power density of 200 W/cm.

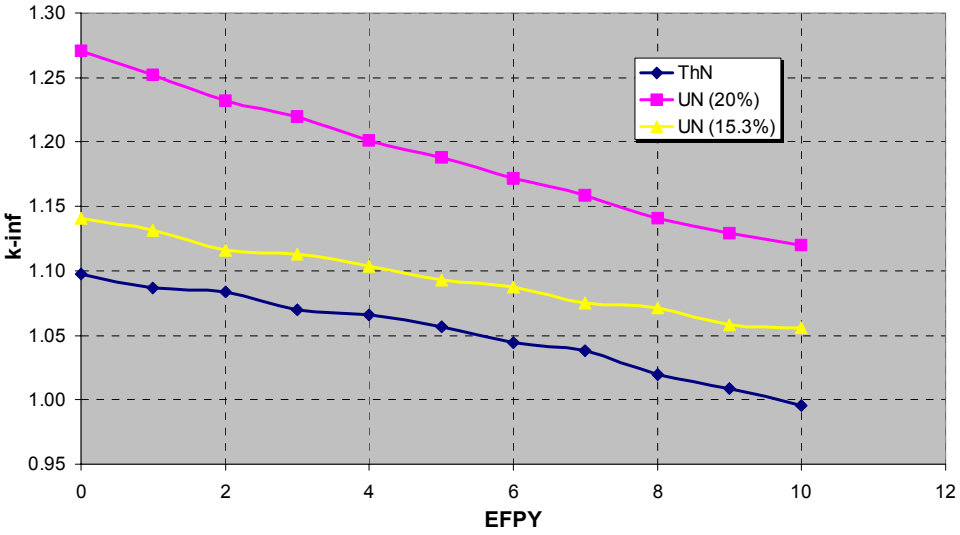

Figure 1. Reactivity versus effective-full-power-years for nitride fuel.

If the end-of-life is chosen where k-infinity = 1.0, then all but one of the fuels will have lifetimes beyond 10 years. The exception is the ThZr fuel, which goes "sub-critical" at approximately 6-7 years. In addition, the lower enriched (i.e., lower weight percent of plutonium and minor actinides) uranium fuels have relatively flat reactivity curves, which are highly desirable from a safety and control aspect. Initially, these lower enriched cases were analyzed in an attempt to lower the beginning-of-life reactivity.

The use of pin cells in such an analysis will not give quantitatively correct results due to the high leakage in the core. However, all of the cases presented here used the same parameters, with the exception of the fuel type. Thus, this qualitative information can be used to select a fuel type, which in turn will be used in full core analyses.

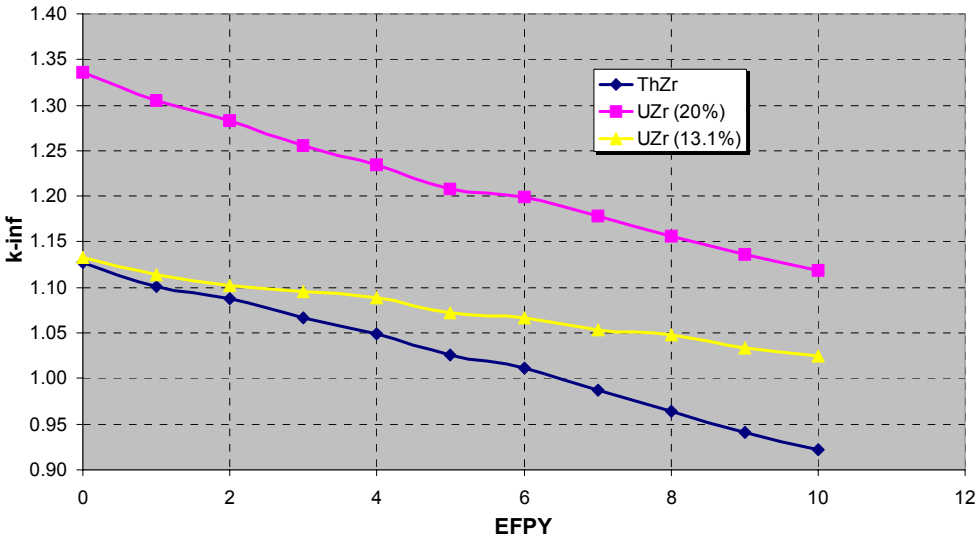

Figure 2. Reactivity versus effective-full-power-years for Zr-metal fuel.

1.3. Isotopic Analysis

The isotopics (plutonium and minor actinides) of the fuel are important; particularly at discharge. This can determine whether the fuel is a good candidate based on proliferation resistance and waste management concerns. Specifically, a total net reduction in initially loaded plutonium and minor actinides is desirable. Figures 3 and 4 give the depletion/generation rates for the specific nuclides of interest. Note that for most of the fuel types, the concentration of the fissile plutonium isotopes decreases whereas the concentration of the Pu-238, Pu-240, and Pu-242 increases, making the waste highly proliferation resistant.

Figure 3. Depletion/generation of nuclides in nitride fuels.

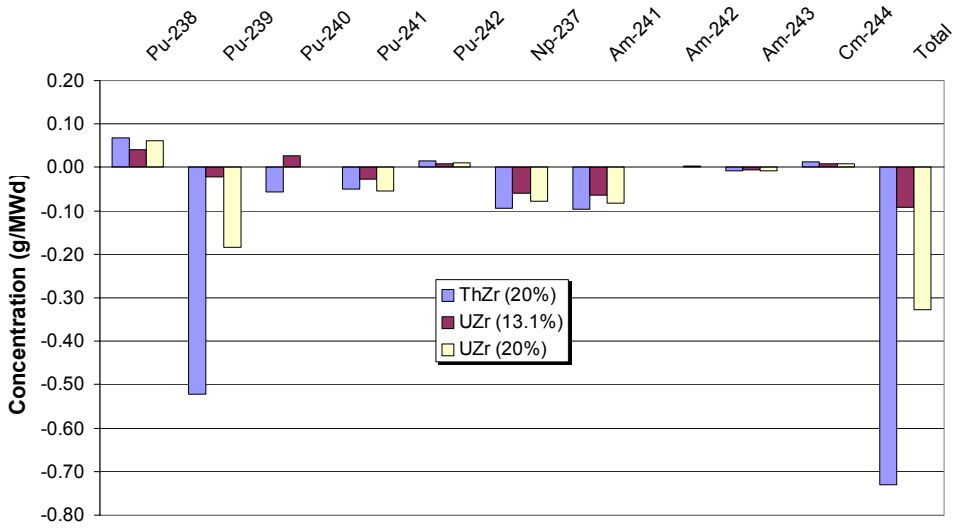

Figure 4. Depletion/generation of nuclides in metallic fuels.

The total net depletion rates are fairly similar for both types of thorium fuel (nitride and metallic), at a rate of about -0.70 g/MWd. The uranium-based fuels have an overall (total) net depletion rate, but it is approximately 2-7 times less than that of the thorium-based fuels. This is due primarily to the additional breeding of plutonium and its associated minor actinides. It is important to note, however, that the thorium fuels will have a significant amount of U-233 present. Addition of some uranium to the thorium fuel will be necessary to denature the U-233, which will affect the performance of this fuel.

Figure 5 shows the total plutonium and minor actinides that are "burned" in a given time period.

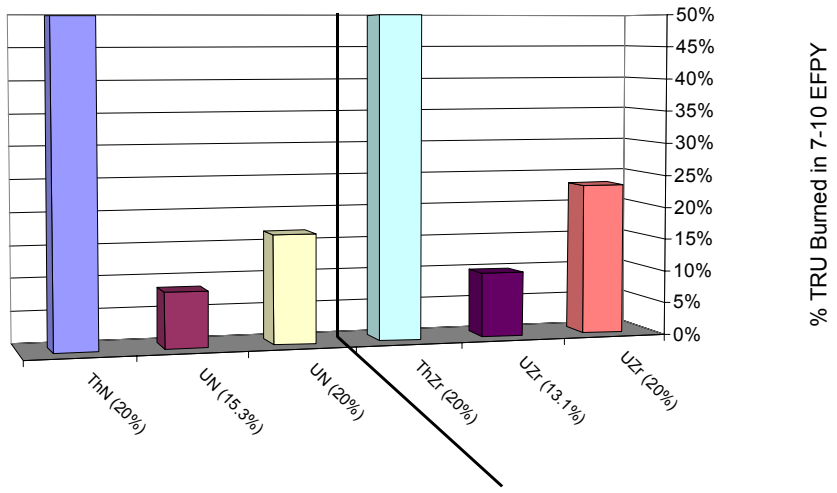

Figure 5. Comparison of plutonium and minor actinide destruction.

Note that the thorium-based fuels will consume about one-half of the plutonium and minor actinides in the fuel.

Also of interest is the plutonium vector (plutonium fractions) at specific points in time. Table 2 shows the plutonium isotopic fractions of each fuel studied.

Table 2. Plutonium vector at specific times.

Fuel	EFPY	Initial Pu+MA	Plutonium Fractions				
			Pu-238	Pu-239	Pu-240	Pu-241	Pu-242
ThN	10	20%	14.4%	25.7%	40.7%	9.8%	9.4%
ThZr	7	20%	14.7%	25.1%	39.6%	10.8%	9.8%
UN	10	15.3%	7.6%	53.5%	28.2%	5.9%	4.8%
UN	10	20%	8.2%	51.0%	29.5%	5.9%	5.3%
UZr	10	13.1%	7.1%	53.1%	28.6%	6.3%	4.9%
UZr	10	20%	8.6%	48.8%	30.6%	6.2%	5.8%

Of interest are the high Pu-238 fractions in the thorium-based fuel (at 14.4% and 14.7%), where Pu-238 has a very high decay heat rate, and spontaneous neutron rate. The high concentration of even-numbered plutonium isotopes for all fuels would make them fairly resistant to proliferation.

1.4. Summary and Conclusions

In this qualitative analysis, four different fuel types were studied, where two of the types had a variation in enrichment. The enrichment used in this study was a combination of plutonium and minor actinides, where the plutonium was typical of that found in LWRs. The uranium-based fuel types had the highest beginning-of-life reactivity, and the best reactivity-limited burnup. However, the thorium-based fuels had the best isotopics, where the net reduction/depletion after 7-10 years was 50%. The most appropriate fuel would have both characteristics, which would appear (from extrapolation) to be a mixture of thorium and uranium to balance long core life with proliferation resistant isotopics.

Task 2. Fuel Cladding and Structural Material Corrosion and Stress Corrosion Cracking Studies

2.1. Progress of Work at MIT

Figure 6 presents a schematic representation of the current SCW facilities at MIT. The exposure facility (for use in this research) incorporates a relatively large autoclave (Figure 7) with an internal volume of approximately 860 mls. It is large enough to expose a rack of samples (weight loss, welded, u-bend) for extended times. The high-pressure liquid chromatograph (HPLC) pump is capable of a maximum flow rate of 100 mls per minute. For previous experiments we have maintained the pre-heater water and corrosive (generally HCl) separate until after the DI water feed is heated to supercritical. The reason for this is that we have observed a correlation between temperature and corrosion rate and mode for our current research, with the worst corrosion appearing to be associated with a high sub-critical temperature. For safety, both facilities are in individual lexan enclosures and control (Labview) is from outside a restricted area. All lines have rupture disks as well as pressure and temperature controllers, and the computer will compensate in the event of a pressure or temperature extreme. Compensation involves turning off the heaters or the pump, or in a very extreme situation the system will be entirely shut down.

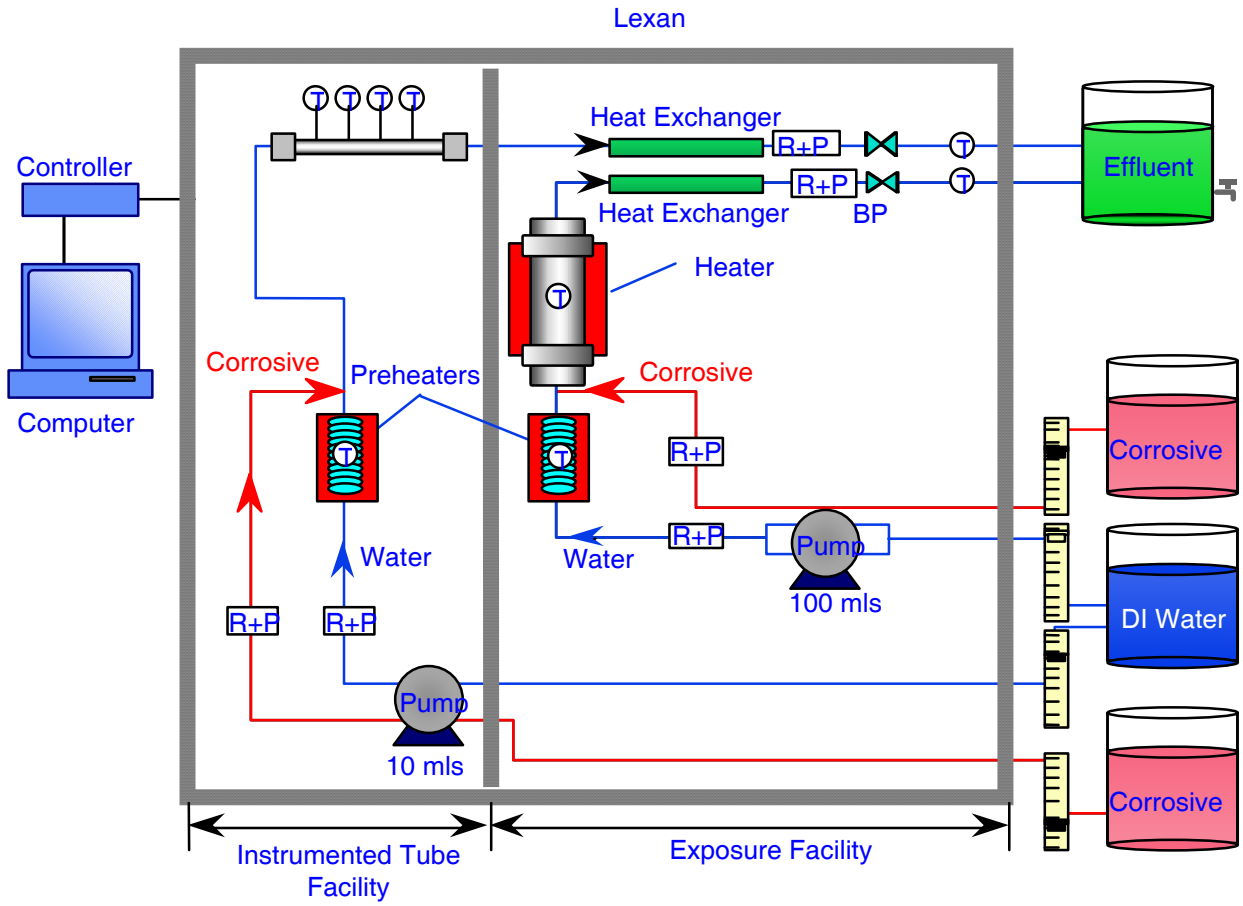

Figure 6. Schematic of the MIT SCW loop.

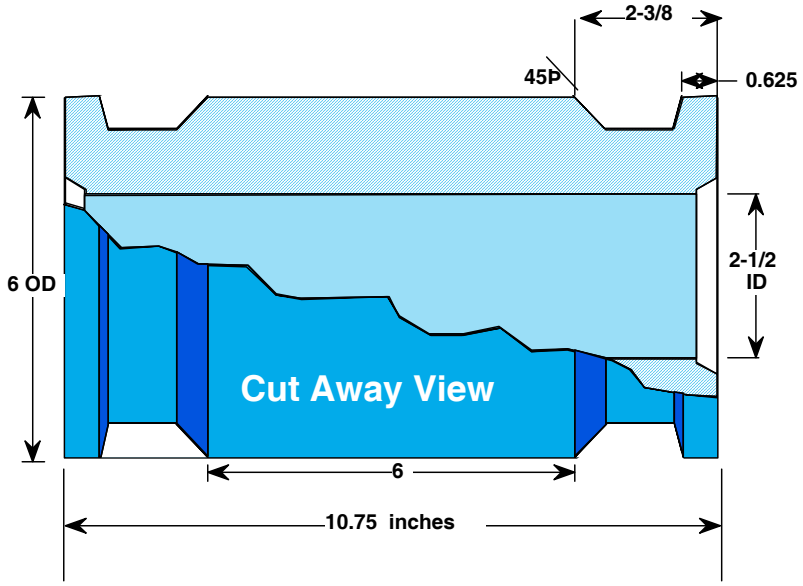

Figure 7. Schematic of the MIT autoclave.

During the first year, the SCW loop at MIT will be employed to carry out initial corrosion experiments on the first set of candidate alloys in flowing supercritical water. The following materials have been identified for testing:

- The focus for the fuel cladding materials will be chiefly on three classes of alloys: ferritic/martensitic stainless steels (e.g., HT-9), austenitic iron-base stainless steels (e.g., 304, 316), and austenitic nickel-base alloys (e.g., Inconel 600 and 690).
- Precipitation-hardened nickel-base alloys (e.g., Alloys 718 and 625) will be considered for the core internals, in addition to the two classes mentioned above.

In order to accomplish the required research, a number of components need to be purchased and modifications to the existing supercritical water oxidation system are necessary. Such modifications are presently in the early stages; however, alterations to the current system should not require significant time. It is, therefore, anticipated that experiments will be initiated in the near future. Once the modifications have been completed, exposure experiments will be carried out in flowing supercritical water. As the internal volume of the autoclave is substantial, relatively large stressed (u-bend) and non-stressed samples of various alloys will be incorporated into the test matrix. Subsequent to an experiment, samples will be metallographically examined. As required, selected samples will be analytically examined. Corrosion rate will be assessed by monitoring mass loss and, of particular interest, samples will be examined for their susceptibility to cracking.

2.2. Progress of Work at the University of Michigan

The design of the supercritical water loop system (SCWLS) for stress corrosion cracking tests was completed and the main components were ordered during the first quarter. In this loop system, one tensile sample can be tested in various loading modes such as constant extension rate tension (CERT), constant load, ramp and hold, low cycle fatigue, etc. Additionally, 6 U-bend samples can be loaded into the test vessel, using sample holders secured to the vessel internal support plate. The system should provide proper test conditions for stress-corrosion-cracking tests such as environmental and loading conditions. The main loop components are the test vessel, loading frame, main pump, heating elements, back pressure regulator, and water columns. Figure 8 shows a schematic of the water loop. The main components and their suppliers are listed in Table 3. Table 4 shows the system performance parameters.

2.2.1. Water Chemistry Control

Water chemistry control includes the control of conductivity, pH, dissolved oxygen content, and the concentration of specific chemicals. The chemistry is controlled in the glass columns. First, the auxiliary column is filled with distilled water from a pure water reservoir. The chemicals required for a test are mixed with the water in the chemistry column, and the mixed water is added to water in the auxiliary column. After the desired water chemistry is achieved, the feed-water is supplied to the main column. Gas cylinders containing argon, nitrogen, or hydrogen are connected to both the auxiliary and main columns. By purging with a gas, the amount of dissolved oxygen in the water is controlled. Water travels from the main pump to the main column and into the test vessel and then back to the main column, establishing a water loop. The main pump controls the flow rate of the circulating water. Water from the test vessel passes the filter and ion exchanger where the corrosion products or any undesired contaminant are removed. During tests, conductivity and dissolved oxygen content are monitored at the inlet and outlet of the test vessel. The water is periodically sampled from the drain line of the main column to measure the pH.

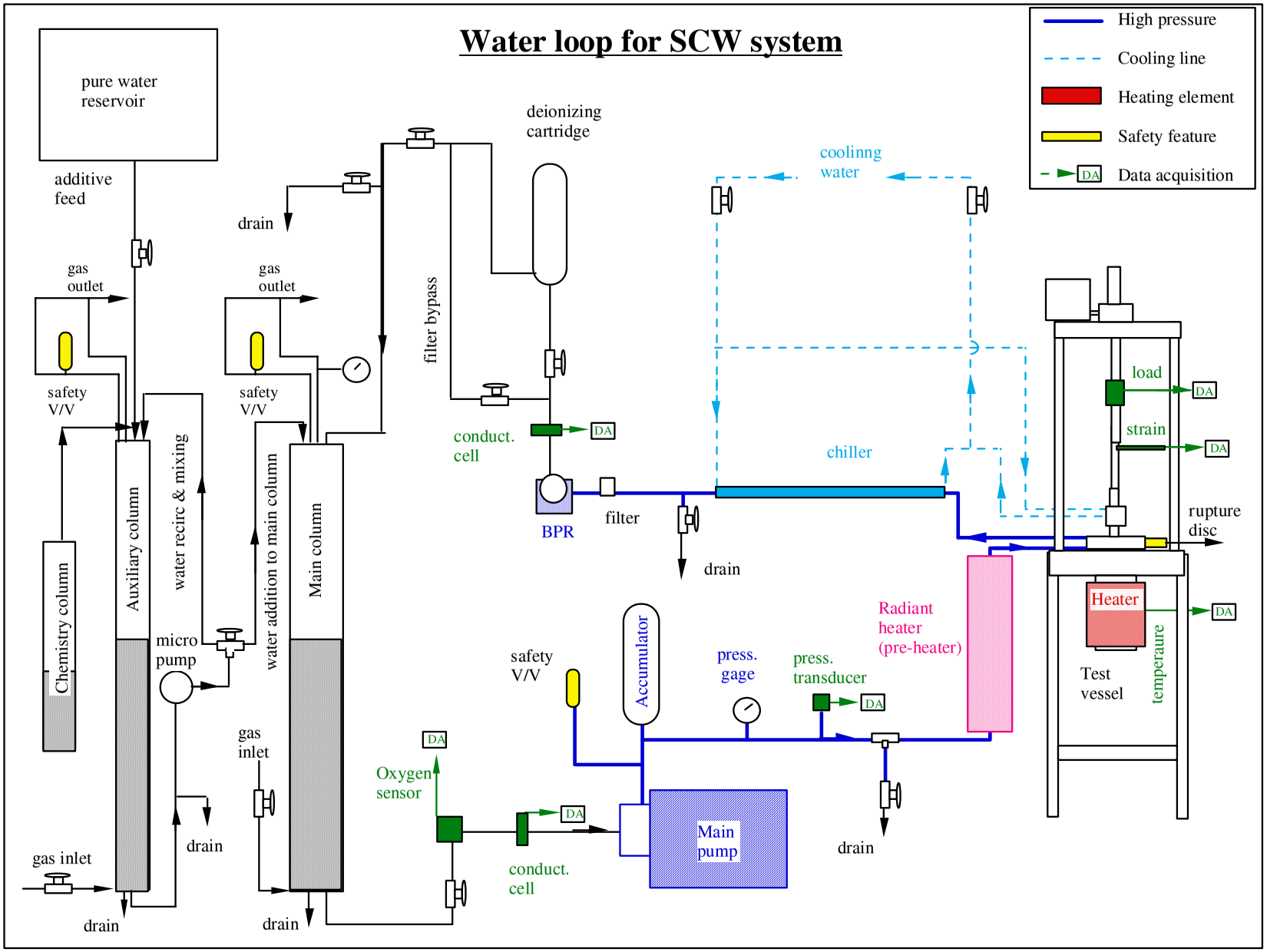

Figure 8. Supercritical Water Loop for stress corrosion cracking tests in the High Temperature Corrosion Laboratory at the University of Michigan.

Table 3. Equipment and suppliers.

Item	Qt'y	Supplier
1. Test Vessel		
Pressure vessel, including heater, internals, and pressure balance system	1 set	Fluid Process Control Corporation
2. Load Frame,		
1) Load frame, including stepper motor, LVDT, load cell	1	Cortest Incorporated
2) Manual screw lift	1	
3. Motor Controller, and Data Acquisition		
1) Computer (data acquisition)	1	M-store (Dell Computer)
2) Motion controller and data acquisition system	1 set	National Instrument
4. Water System		
1) Oxygen meter	1	Orbisphere
2) 2-cell conductivity meter	2	Omega/ CP
3) HP water filter for 2nd.+case	2	Fisher
4) Fittings, tubing, and valve		HE Lennon
5. Heating System		
1) Ceramic radiant heaters	1	Omega
2) PID temperature controller and meter, thermocouple	2 set	Omega
6. Pressurization System		
1) High pressure pump and parts/accessory kit	1 set	Eldex
2) Pressure transducer	1	Omega
3) Back-pressure regulator and repair kit.	1 set	Accu Flo
4) Pressure gauge	1	Omega
5) Pressure accumulator	1	Hydac
7. Miscellaneous		
1) Electronics supplies	1	Newark
2) Misc. construction supplies		Home Depot

Table 4. System capabilities and monitoring.

Condition	Range	Equipment	Remarks
-Pressure	Max. 3450 psig (25MPa)	Triple piston pump and back-pressure regulator	Continuous monitoring
-Temperature	Max. 500°C	Main heater and radiant pre-heater	Continuous monitoring
-Conductivity	Higher than 15MW-cm	Filter and ion exchanger	Continuous, monitoring
-Dissolved oxygen	Below 10ppb - 2ppm	Argon gas	Continuous monitoring
-PH			Discontinuous
-Mechanical loading mode	Constant extension	Stepper motor and motion controller	Sample load and extension monitored continuously
	Constant load		
	Step extension		
	Step load		

2.2.2. Supercritical Water Condition

The state of water is determined by the temperature and pressure. Water circulated by the main pump is pressurized via the back-pressure regulator. A small pin in the back-pressure regulator controls the amount of water flowing out of the back-pressure regulator, maintaining a constant pressure. The pressure control is achieved by rotating the handle on the back-pressure regulator and checking the pressure indicated by the pressure gauge. The accumulator installed at the outlet of the pump reduces the fluctuation of the pressure by the pump. The water is heated to a certain temperature by the radiant pre-heater before it flows into the vessel, where it is heated up to the desired temperature by the main ceramic heater. The water flowing out of the vessel is cooled in the double tube chiller. During tests, pressure and temperature in the vessel are monitored and recorded by the pressure transducer connected to the tubing of the pump outlet, and by the thermo-couple secured inside of the vessel, respectively.

2.2.3. Mechanical Loading

Once the environmental conditions, water chemistry, pressure, and temperature are achieved, a load can be placed on the tensile sample. Mechanical loading will be achieved by the stepper motor attached to the load frame. A tensile sample is connected to the motor through a pull rod on which a linear voltage displacement transducer (LVDT) and a load cell are installed. The stepper motor is controlled by a motion controller. The LVDT and load cell monitor the amount of strain and load on a sample. The loading mode can be programmed using LabView software connected to the motion controller. The motion controller controls the motor function using the input values of strain and load and programmed loading mode. The rotation of the motor determines the displacement of the pull rod that in turn determines the mechanical condition of a sample.

One unique feature of this system is that a pressure balance system is attached to the top of the test vessel. Without the pressure balance system, when the water in the vessel is pressurized, the pressure would push the pull rod outward, which would result in undesired pre-straining of a sample. The high pressure of a supercritical water system would impose a strain load on the sample high enough to cause plastic deformation. On the other hand, the pressure balance system maintains the strain (displacement) of a sample constant regardless of the water pressure before loading.

2.2.4. Safety Features

Since the loop is operated at high temperature and pressure, some safety features are required. There are two safety features on the high-pressure section between the pump and back-pressure regulator. These are a safety valve and a rupture disc. The safety valve is located just after the pump and is designed to open automatically when the water pressure increases beyond a set value. The disc is designed to endure the design pressure and will rupture at higher pressure. Also, the auxiliary and main columns each have their own safety relief valves.

Task 3. Plant Engineering and Reactor Safety Analysis

3.1. Progress of Work at the INEEL

In preparation for the transient and accident analysis that will be conducted at the INEEL in Year 2 and 3 of the project, the 3D finite-differences transient thermal-hydraulic/neutronic code, RELAP-ATHENA was upgraded.

The RELAP5 (INEEL 2001) computer program has been extensively used and validated for the thermal-hydraulic analysis of light water reactors over the last 20 years. However, nearly all of the applications have been performed at pressures less than the critical pressure, consistent with the design and normal operation of current light water reactors. The code contains fluid properties for supercritical light water, but has been used successfully for only a few supercritical cases (Buongiorno and Davis 2001). Experience has shown that the code does not execute as reliably at supercritical conditions as it does for sub-critical conditions.

A simple test problem was simulated to evaluate potential code problems for supercritical applications. The test problem represented a pipe that initially contained sub-cooled water at less than critical pressure. Boundary conditions were varied so that the pressure in the pipe increased along a line of constant temperature until reaching a maximum value that exceeded the critical pressure of 22.12 MPa. The pressure was then held constant at the maximum value and the temperature increased until it was above the critical temperature of 647.3 K. The pressure and temperature were then decreased back below the critical values. The initial pressure and temperature were 0.4 MPa and 322 K, respectively. The final state corresponded to superheated vapor at a pressure of 5 MPa.

Twenty-seven cases were run in which the maximum pressure varied from 22.2 to 90 MPa. The loci of points of calculated pressure versus temperature are shown in Figures 9 through 11 for the 27 cases. Each figure contains a dashed line that represents the saturation curve as a function of temperature. The transient began in the lower left corner of the graph and ended in the lower right corner. If a code failure was obtained during the calculation, the locus of points does not return to 5 MPa in the lower right corner of the graph. Figure 9 indicates that two of eight cases failed when the maximum pressure was between 25 and 90 MPa. Figure 10, which contains results from 10 cases with a maximum pressure between 23.1

and 24 MPa, shows that only one case completed successfully. Figure 11 shows that all of the calculations failed when the maximum pressure was between 22.2 and 23 MPa. Overall, 20 of the 27 cases failed with the original code, and the probability of failure increased dramatically near the critical point.

Several code updates were implemented to correct the code execution failures indicated in Figures 9 through 11. Specifically, the steam tables were changed to use more consistent values for the reset water property derivatives representing the specific heat and the coefficients of volume expansion and isothermal compressibility at the critical point. The steam tables were also generated using more pressure and temperature points near the critical point. The specific volume and isothermal compressibility were interpolated using linear, rather than cubic, expressions. Changes were also made to the extrapolations for metastable states near the critical point. Additional protections were provided to prevent problems with square roots and log functions. The calculations described previously were repeated with the updated code. All 27 cases executed successfully with the updated code as illustrated in Figures 12 through 14.

The updated code is now suitable for initial analysis of the supercritical reactor. Additional code problems will probably be encountered, particularly during transients that pass near the critical point because of the steep slopes in thermodynamic properties. These potential problems will be addressed as needed.

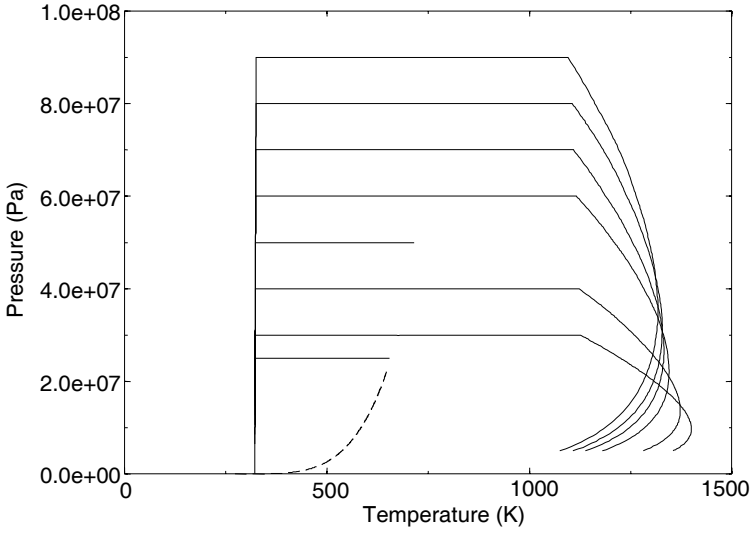

Figure 9. Pressure-temperature plots for cases with the maximum pressure between 25 and 90 MPa (no updates).

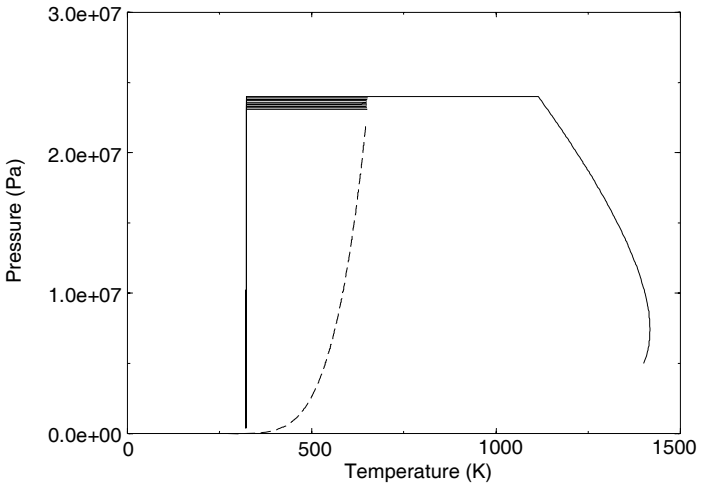

Figure 10. Pressure-temperature plots for cases with the maximum pressure between 23.1 and 24 MPa (no updates).

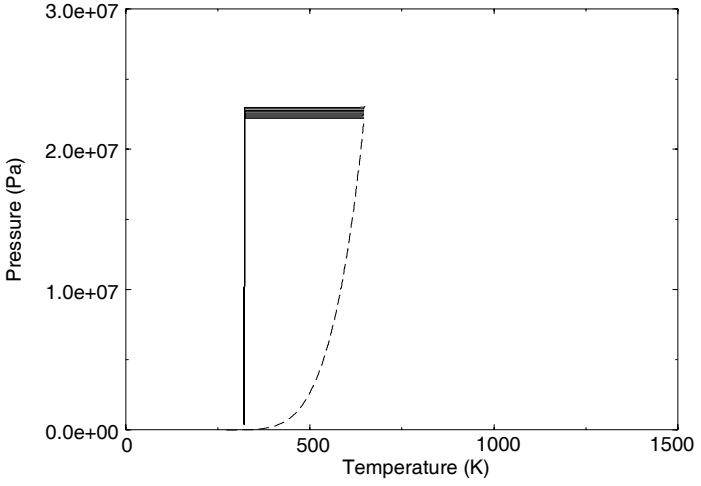

Figure 11. Pressure-temperature plots for cases with the maximum pressure between 22.2 and 23 MPa (no updates).

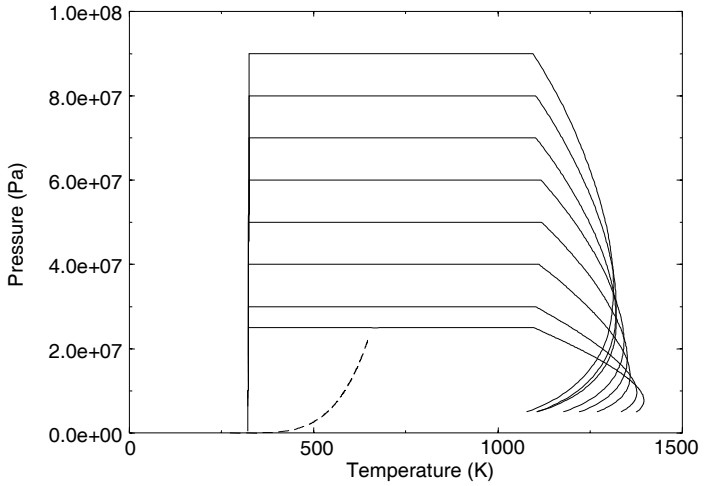

Figure 12. Pressure-temperature plots for cases with the maximum pressure between 25 and 90 MPa (with updates).

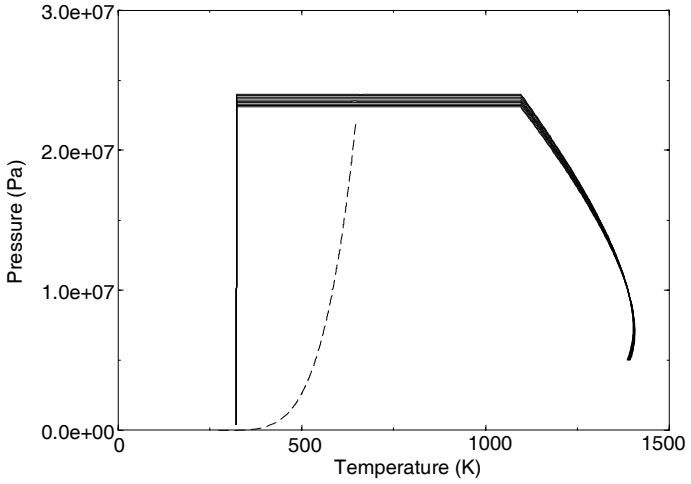

Figure 13. Pressure-temperature plots for cases with the maximum pressure between 23.1 and 24 MPa (with updates).

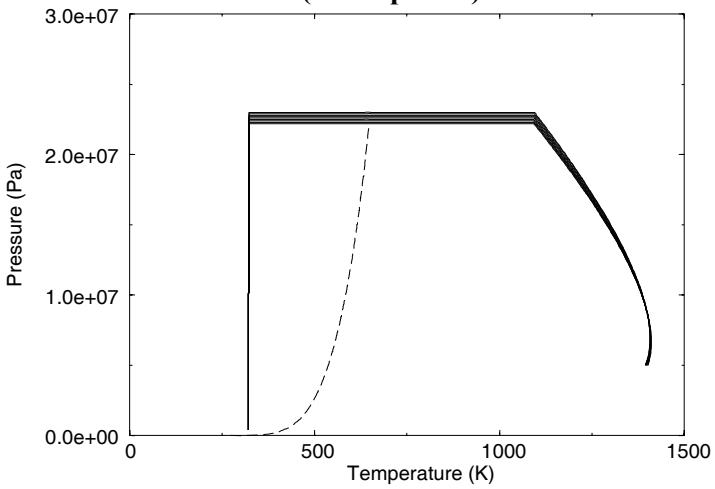

Figure 14. Pressure-temperature plots for cases with the maximum pressure between 22.2 and 23 MPa (with updates).

3.1. Progress of Work at Westinghouse

The INEEL was unable to execute a subcontract with the Westinghouse Electric Co. for work on this NERI project because Westinghouse does not have a DOE approved financial system and because Westinghouse as refused to accept the INEEL's standard contract conditions. We expect this problem to be solved in the second quarter of this project.

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Project Schedule

Task	Activity	Description	Year 1	Year 2	Year 3
Task 1		Fuel-cycle Neutronic Analysis and Reactor Core Design (INEEL)			
	1.1	Reactivity Swing Analysis			
	1.2	Actinide Discharge and Isotopic Evaluation			
	1.3	Reactivity Coefficient Calculations			
	1.4	Peaking Factors and Reactor Control			
Task 2		Fuel Cladding and Structural Material Corrosion and Stress Corrosion Cracking Studies (University of Michigan, MIT)			
	2.1	Identification of Most Promising Materials (MIT)			
	2.2	Design and Construction of an Out-of-pile Supercritical Water Test Facility (U-Mich)			
	2.3	Corrosion and Stress Corrosion Cracking Behavior of Candidate Materials (U-Mich)			
	2.4	Radiation Stability of Candidate Alloys (U-Mich)			
	2.5	Modeling of Corrosion and stress Corrosion Cracking in Supercritical Water (U-Mich)			
Task 3		Plant Engineering and Reactor Safety Analysis (Westinghouse and INEEL)			
	3.1	Conceptual Design of the Reactor Coolant System (Westinghouse)			
	3.2	Definition of the Thermal/Mechanical Design Limits			
	3.3	Core Thermal-hydraulic Design (Westinghouse)			
	3.4	Evaluation of Coupled Thermal-hydraulic/Neutronic Oscillations (INEEL)			
	3.5	Plant Configuration and Operation (Westinghouse)			
	3.6	Establish the Conceptual Design of Required Safety Systems and Define their Performance Parameters (Westinghouse)			
	3.7	Analysis of Anticipated Transients and Potential Accidents (INEEL)			
	3.8	Conceptual Layout of Reactor Containment, Fuel Handling, and Auxiliary Buildings (Westinghouse)			
	3.9	Economic Analysis (Westinghouse)			

Budget and Actuals for Year 1

Organization	Year 1 (Budget)	Year 1 (Actuals, Sep01-Dec01)
INEEL	100.0K	4.0K
University of Michigan	140.8K	0.0K*
MIT	48.5K	0.0K*
Westinghouse	96.1K	0.0K**
Total	385.4K	4.0K

* Because of administrative problems in finalizing the contracts, the University of Michigan and MIT started their work on this NERI project with internal funds.
** Because of administrative problems in finalizing the contract Westinghouse has not started working on this NERI project.

INEEL/EXT-02-00107

Feasibility Study of Supercritical Light Water Cooled Fast Reactors for Actinide Burning and Electric Power Production

Nuclear Energy Research Initiative Project 2001-001

Progress Report for Year 1, Quarter 1 (Sep 01 - Dec 01)

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