INEEL/EXT-02-00759

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 2
(January through March 2002)

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Idaho National Engineering and Environmental Laboratory
Bechtel BWXT Idaho, LLC

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Collaborating Organizations:

Massachusetts Institute of Technology
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University of Michigan
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Table of Contents

Project Description

The use of light water at supercritical pressures as the coolant in a nuclear reactor offers the 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 about 45%). 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 and smaller containment building 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 supercritical water-cooled reactor (SCWR). This type of core can make use of either fertile or fertile-free fuel and retain a hard spectrum to effectively burn plutonium and minor actinides from LWR spent fuel while efficiently generating electricity. One can also add moderation and design a thermal spectrum SCWR that can also burn actinides. The Generation IV Roadmap effort has identified the thermal spectrum SCWR (followed by the fast spectrum SCWR) as one of the advanced concepts that should be developed for future use. Therefore, the work in this NERI project is addressing both types of SCWRs.

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

1) Fuel and Reactor Core Designs:

- Local or total coolant voiding in the fast-spectrum SCWRs 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.
- The thermal-spectrum SCWRs require additional moderation, water rods can be used but one has difficult design problems to control the heat transfer from the coolant to the moderator rods, especially during off-normal and accident situations. A solid moderator would be better.
- 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.
- The Doppler feedback in the fast-spectrum SCWRs 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.
- Radiolysis of the water coolant in the fast-spectrum SCWRs may take place at a higher rate than in traditional LWRs. In addition, the radicals formed by the radiolytic decomposition of the water (both fast and thermal versions) are highly soluble in supercritical water and may not recombine as well as in an LWR.
- The hard neutron spectrum in the fast-spectrum SCWRs 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 relative benefits of direct versus indirect cycle reactor coolant system designs need to be assessed.

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

Task 1. Fuel-cycle Neutronic Analysis and Reactor Core Design (INEEL). For the fast-spectrum SCWR, 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. The use of thorium will be included in the fertile options. The main variables are the core geometry (e.g. fuel rod length, pitch-to-diameter ratio, assembly configuration) and the fuel composition. For the thermal-spectrum SCWR, a variety of fuel and moderator types will be assessed. 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 (January through March 2002)

Because of delays in subcontracting, this NERI project started at the INEEL MIT, and the University of Michigan at the beginning of November 2001. The INEEL was unable to get Westinghouse to accept a subcontract, so the work at Westinghouse did not start until a contract from DOE-OK was placed in March, 2002.

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

1.1. Summary of Previous Work

During the 1st quarter, a qualitative analysis was performed to determine which fuel form would support the highest reactivity-limited burnup in the fast-spectrum SCWRs, and would have the most proliferation resistant isotopics at a particular burnup. About 13-20wt% of the fuel was plutonium and minor actinides, with the remainder of the fuel consisting of uranium or thorium as either mono-nitrides or in a zirconium-metal matrix.

Figure 1 shows the reactivity versus the effective-full-power-years (EFPY) for the metallic fuel. Note that relatively long core life and a modest reactivity swing are possible in fast-spectrum SCWRs. 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 spent fuel 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.

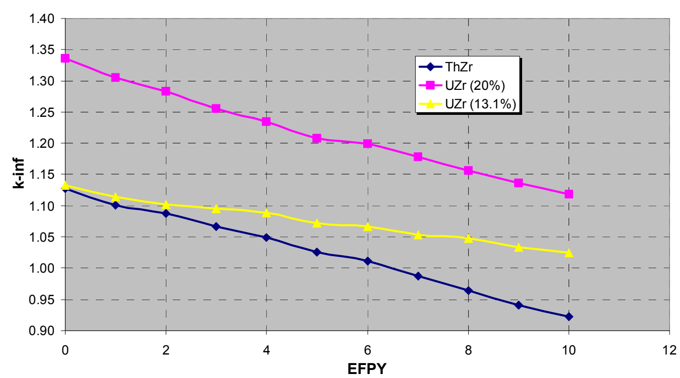

Figure 1. Reactivity versus effective-full-power-years for Zr-metal fuel.

1.2. Introduction to the 2nd Quarter Work

It is well known that the fission to capture cross-section ratio (σ_f/σ_c) increases for the transuranics in a fast neutron spectrum, making fast spectrum systems attractive for plutonium and minor actinide management. However, the small effective delayed neutron fraction (β_{eff}) associated with these systems can make reactor control problematic, especially for fuels that contain large quantities of minor actinides. To help overcome this challenge, and remain within the current knowledge of thermal spectrum reactor control, the work done for this quarter has included parametric studies of thermal-spectrum, supercritical pressure water pin cells. While this may seem to contradict the goal of actinide management within this reactor concept, the capture cross-section of minor actinides in a thermal spectrum is large. The minor actinides can be rapidly transmuted by neutron capture to fissile isotopes with high fission cross sections, and are eventually destroyed. More importantly, the thermal spectrum cross-sections for neutron absorption are 200-300 times larger than for fast neutrons.

On the other hand, the average axial coolant density across the core is quite small, which necessitates an addition of moderator to thermalize the neutrons. In the work presented here, supplementary moderator was added by increasing the pin pitch, and utilizing rods that contained several different moderators. A comparison was made between these two approaches, where a primary reference case was chosen that contained the same hydrogen (moderator) to heavy metal ratio (H/HM) as in a standard 17x17 PWR lattice (but with a much larger pitch so as to have enough moderator). A secondary reference case contained water rods at a constant density, and the pin pitch was decreased until the same H/HM as the primary reference case was achieved. All other moderators used the secondary reference case parameters. The main purpose of this study was to generate data that will help in choosing an appropriate moderator for thermal-spectrum SCWRs.

1.3. Fuel Parameters and Analysis Tools

The parameters of the pin cells containing moderator rods can be found in Table 1. Note that the fuel used was a 5% enriched (U-235) UO_2, and that a square pitch lattice is used. The primary reference case used the same parameters, with the exception of the pin pitch, which was 1.43 cm. In addition, the moderator rods also used the same parameters as the fuel rods. A diagram of the unit cell used in the calculations is shown in Figure 2. The cladding was assumed to be E911, a ferritic stainless steel with 9% chromium.

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)	366
Pin Pitch - square (cm)	0.95
Fuel Temperature - average (K)	900
Number of Coolant Nodes	22

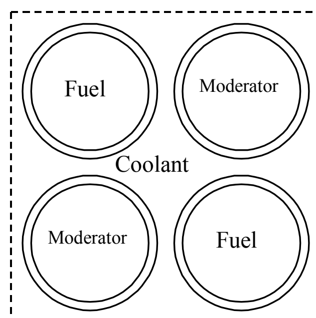

Figure 2. Unit cell representation for fuel and moderator rods.

MCNP (Briesmeister 1997), a well-known Monte Carlo code, was used to calculate the beginning-of-life (BOL), infinite neutron multiplication factor (k_{inf}), and the neutron spectrum of each case.

1.4. Moderating Power and k-Infinity

Initial calculations were performed to verify the moderating power and moderating ratio of several different moderators. The moderators in the rods included H_2O, D_2O, graphite (C), BeO, and ZrH_2. Note that not all of the moderators chosen would be compatible in this reactor environment, but were still used as comparisons. Table 2 summarizes the moderating capability of these moderators.

Table 2. Moderating parameters.

Moderator	ξ	ρ	σ_s	σ_a	Σ_s	Σ_a	$\xi\Sigma_s$	$\xi\Sigma_s/\Sigma_a$
H_2O	0.92	0.705	103	0.664	4.855	0.0313	4.466	142.711
D_2O	0.509	0.776	13.6	0.001	0.634	0.0001	0.323	5204.812
C	0.158	1.8	4.75	0.003	0.429	0.0003	0.068	220.413
BeO	0.207	3.01	6.14	0.009	0.445	0.0007	0.092	137.884
ZrH_2	0.92*	5.2	103*	0.848	6.918	0.0570	6.365	111.745

*Estimated value.

The last two columns in Table 2 show the calculated moderating power and moderating ratio, respectively, where ξ is the mean lethargy gain per collision, and is only dependent on the atomic mass of the moderating nuclei. From this table, we can see that H_2O and ZrH_2 appear to be the best moderators, while BeO and graphite would be the worst performers. However, the k_{inf} calculations do not necessarily show exactly the same trend. A comparison of the beginnning-of-life k_{inf} can be seen in Figure 3.

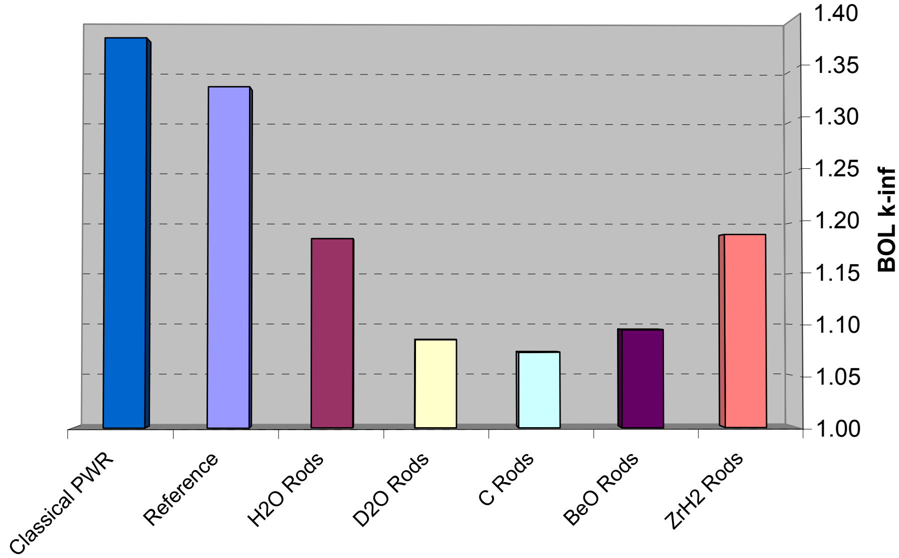

Figure 3. Comparison of beginning-of-life k_{inf} for each moderator.

Of particular interest is the large difference between the primary reference case and the H_2O moderator rod case. These two cases contained the same H/HM ratio, although the moderator rod case used a much tighter pin pitch. Recall, however, that in calculating the H/HM ratio, the average axial coolant density was used rather than the explicit density at each node, resulting in an average H/HM ratio (or equivalent moderator to fuel ratio for other moderators). Because the volume of the coolant in the reference case is almost 5 times that of the moderator rod case, and the entry plane density of the coolant is high, the bottom portion of the pin can be viewed as over moderated which compensates for the under moderated top portion. This results in a spectrum and reactivity that is comparable to a classic, 17x17 PWR unit cell. In addition, the very large pitch of the reference case would not be feasible in this reactor due to the resultant low coolant velocities and high cladding temperatures. Nonetheless, the results of the remaining moderator rod cases can be compared to the H_2O moderator rod case, where the ZrH_2 is slightly better than the H_2O, and the BeO follows as the next best performer.

1.5. Spectral Effects

Based on the previous discussion of moderating power and ratio, one would have expected the D_2O to perform better than the BeO. However, spectral effects play a major role in the reactivity and burnup capability of the fuel. More thermalized spectrums, up to a point, will result in higher burnups and higher beginning-of-life reactivities. The beginning-of-life k_{inf}'s from the previous section can be correlated directly to the spectrum, as can be seen in Figure 4.

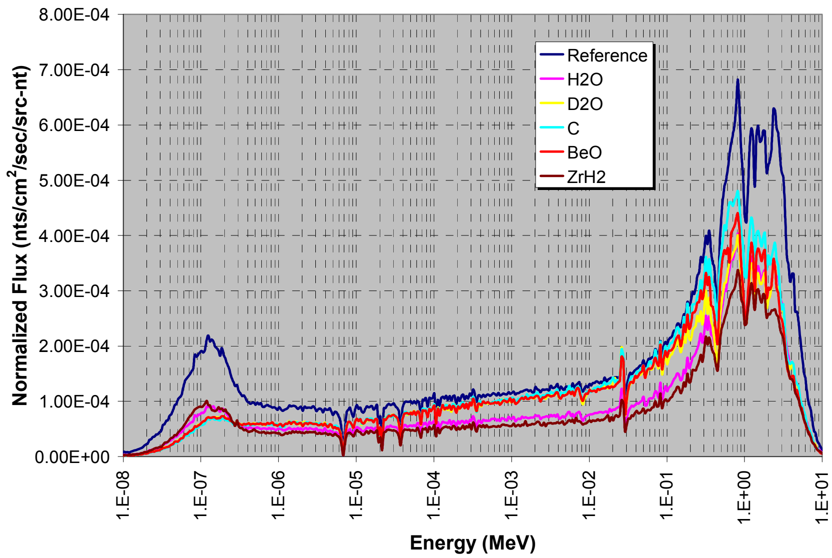

Figure 4. Neutron energy spectrums for the reference case, and the moderator rod cases.

Note the high thermal spectrum of the reference case, and the lower but similar thermal spectrums of the H_2O and ZrH_2 cases. Again, the difference in coolant volumes accounts for the difference in spectrum, and, therefore, beginning-of-life k_{inf}.

1.6. Conclusions

In order to thermalize the neutron spectrum in a supercritical pressure water reactor, one can increase the pin pitch, and thus increase the moderator to fuel ratio, or introduce moderator rods (or cans) to increase the moderator to fuel ratio. While the former concept would be the simplest, the large hydraulic diameter would result in lower coolant velocities and unacceptable cladding temperatures. Therefore, one must employ moderator rods/cans that can be used to increase the moderator to fuel ratio.

The work presented here gives a qualitative indication of which moderators will perform best for the given parameters, where ZrH_2 and H_2O outperform the other moderators, with BeO being the next best performer. However, due to the exothermic reaction with water, ZrH_2 would most likely be eliminated as a moderator in this system. The remaining moderators under consideration (H_2O and BeO) will need further evaluation based on design practicality and thermal performance indicators.

1.7. Future Work

Future calculations will involve the refinement of the moderator to fuel ratios for more exact comparisons by using the explicit densities at each node, and several moderator rod/can designs will be used to maximize both the thermal-hydraulic performance and moderator to fuel ratio. In addition, fuel containing transuranics (TRU) will be studied for both burnup potential, and TRU destruction rates as compared to the fast spectrum cases.

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

2.1. Progress of Work at MIT

The investigators from the University of Michigan and MIT met in Ann Arbor on February 13th and 14th. The primary purpose of the meeting was to discuss the capabilities of the individual systems, and to define the experimental conditions and materials to be evaluated during the initial experiments. MIT will host a follow-up meeting on 28 June in Cambridge.

2.1.1. Identification of Most Promising Materials

At the culmination of the meeting in Ann Arbor, it was decided that initial experiments should be carried out using 316L, as a baseline material, and I-625. Preparations are currently underway to test these materials in 15-mega ohm type water over a temperature range encompassing both sub- and supercritical conditions. At the same time, the availability of a broad range of materials (see Tables 3 and 4 in the University of Michigan section below) in tube form is being pursued. A formal literature survey is underway, but is being carried out by a postdoctoral student who has only recently joined the Corrosion Laboratory at MIT. A report on the literature is anticipated at the time of the next reporting cycle.

2.1.2. Corrosion and Stress Corrosion Cracking of Candidate Materials

Figure 5 presents a schematic representation of the current SCW facilities at MIT. The exposure facility (for use in this research) incorporates a relatively large autoclave 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 our previous waste treatment studies 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

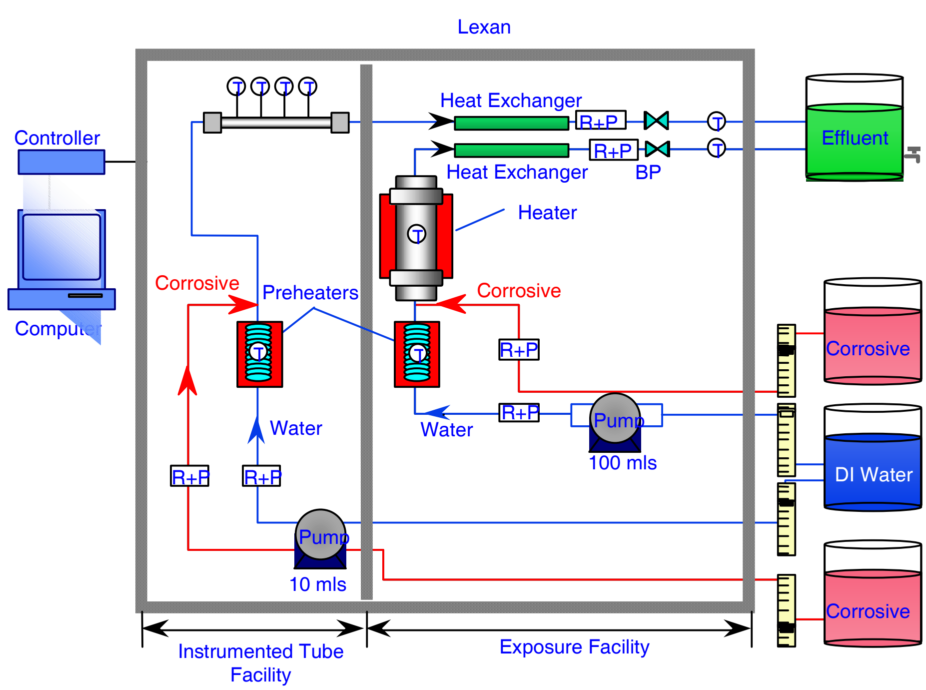

Figure 5. Schematic of the MIT SCW loop.

enclosures and control (Labview) is from outside a restricted area.

In the exposure facility, mass loss coupons and u-bend samples will be tested for extended times. The sample rack shown in Figure 6 was designed to maximize utilization of the vessel volume (≈ 864 cm^3), while not restricting fluid flow. As shown in Figure 7, the rack and samples are sealed into an I-625 exposure vessel during an experiment. Subsequent to a test, samples will be examined metallographically and analytically to assess corrosion rate and mode. While only one temperature and feed condition may be tested during each individual exposure, multiple materials can be evaluated.

Figure 6. The sample rack that will be used during the MITexposure studies. The samples are electrically isolated from each other and the rack by means of alumina spacers.

Previous research at MIT has indicated a correlation between temperature and degradation for SCW systems; therefore, it is considered important that potential materials should be assessed in the most aggressive temperature range. In order to define this temperature range, initial testing will be accomplished with the instrumented autoclave design, which has successfully been employed in the Uhlig Laboratory during previous work. This design permits simultaneous exposure of a single material (a tube or a wire) to multiple temperatures. A photograph of the system configured for this type of work is presented in Figure 8, and a schematic of the autoclave is presented as an insert. The feed is heated to the temperature of interest by pumping it through a 20-foot pre-heater located just prior to the autoclave. Thermocouples are spaced equidistantly (insert), and the temperature decreases along the length of the autoclave, with the highest temperature at the inlet. The output from each thermocouple is monitored and recorded throughout the test, and, thus, the corrosion mode and extent can ultimately be correlated to temperature. The fluid flows out of the top of the autoclave and is directed through a cool down heat exchanger before passing through the back-pressure regulator (BPR) and into a collection carboy.

Figure 7. The rack of samples is inserted into the Inconel 625 exposure vessel prior to an experiment.

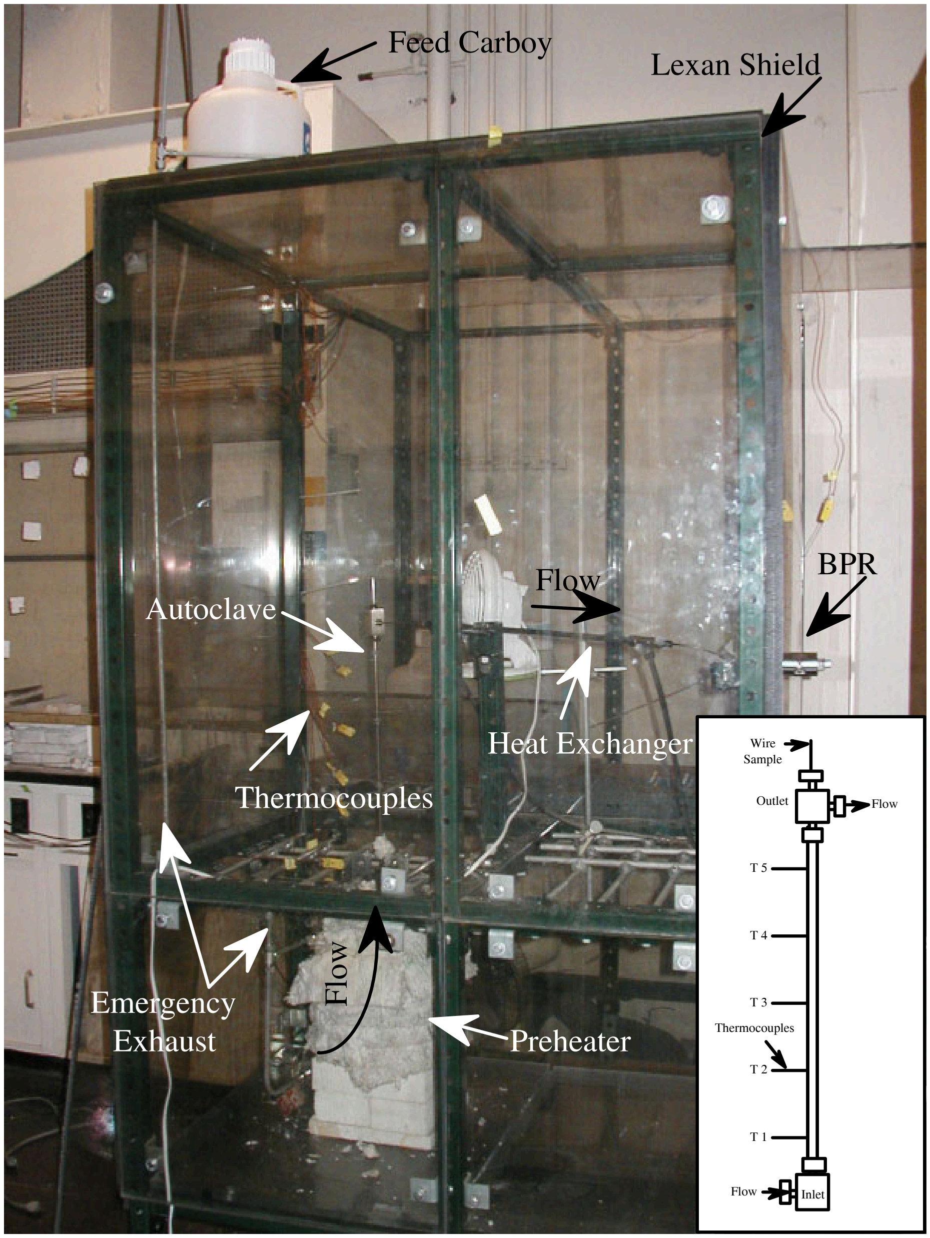

Figure 8. The Inconel 625 instrumented autoclave system in the MIT SCW laboratory.

2.2. Progress of Work at the University of Michigan

During the second quarter a meeting was held between MIT and the University of Michigan to discuss candidate alloys and corrosion and SCC (stress corrosion cracking) test conditions. Also, based on the preliminary design of the supercritical water loop system (SCWLS), general and specific requirements for the main components including load frame and test vessel were determined and the main components were ordered. A second meeting will occur on June 28, 2002 at MIT, immediately after the Generation IV meeting in Boston that week.

During the quarter the construction of the SCWLS was begun.

2.2.1 Identification of Most Promising Materials

Based on literature and previous experience, various alloys for SCWRs were discussed by MIT and the University of Michigan and candidate alloys for testing were determined at the meeting on the 13th and 14th of February between both the parties. MIT has the lead on this subtask, and several of these issues were reviewed at the meeting. In the discussion, application history to fossil plants, alloy limitations, properties of alloys and considerations for potential application to nuclear plants were also considered. The alloys reviewed at the meeting are listed in Table 3 and the chemical compositions of the alloys are shown in Table 4. In Table 3, the alloys shown in bold letters were selected as the highest priority candidate alloys. It was planned that MIT would perform a set of preliminary test using tubes of the candidate alloys and based on the corrosion results, the stress corrosion cracking (SCC) test plan would be made.

2.2.2 Design and Construction of an Out-of-pile Supercritical Water Test Facility

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 9 shows a schematic of the water loop.

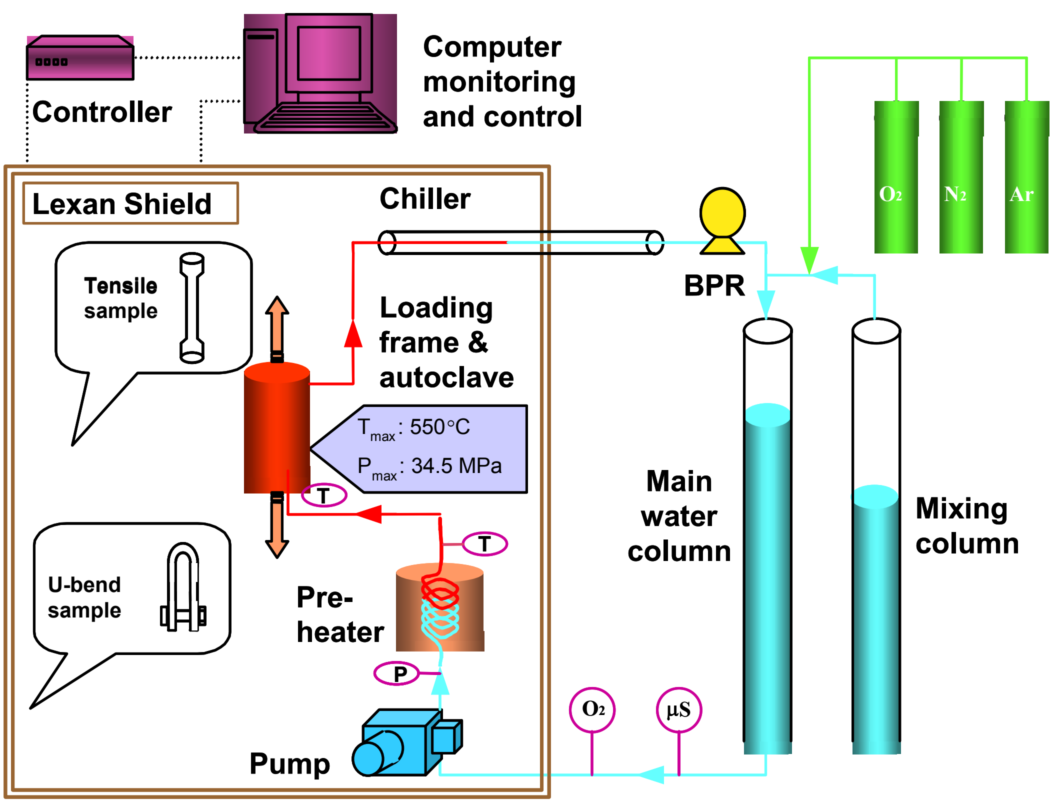

Figure 9. Supercritical Water Loop for stress corrosion cracking tests in the High Temperature Corrosion Laboratory at the University of Michigan.

Table 3. Candidate alloys for supercritical water reactor core materials application

Class	Alloy	Application history	Remarks
Austenitic stainless steel	Stainless steel 304	- Reactor internals in LWRs	- Good general corrosion resistance and moderated strength through the 400-500°C range. - Loss of strength at the upper end of the range, localized corrosion (IGSCC) in oxidizing conditions and susceptibility to radiation-induced swelling in the 400-600C range.
	Stainless steel 316	- Reactor internals in LWRs - Cladding in LMFBRs	
Solid solution Ni-base austenitic alloy	Alloy 600	- Steam generator tubing in PWRs	- Susceptible to IGSCC in PWR conditions.
	Alloy 690		- Less susceptible to IGSCC than alloy 600.
Precipitation hardened Ni-base alloy	**Alloy 625**	- Reactor-core and control rod components in LWRs	- Hardened by γ" phase [Ni_3(Nb,Al,Ti)] precipitated by aging. - Higher strength up to about 700°C than solid solution Ni-base alloys.
	Alloy 718	- Reactor core internals in LWRs	- Hardened by γ" [Ni_3(Nb)] and γ" [Ni_3(Nb,Al,Ti)] precipitates. - Higher strength up to about 700°C than solid solution Ni-base alloys.
Ferritic/martensitic steels	HT-9	- Structural applications in supercritical power plants	- Lower coefficient of thermal expansion and higher thermal conductivity than austenitic steels. - High swelling resistance in fast neutron, ion, and electron irradiation conditions.
	T-91		
	HCM12A		

Table 4. Nominal composition of candidate alloys

Alloy	Cr	Ni	Fe	C	Mo	Cu	W	Ta	Nb	Al	Ti	Mn	Si	V	N	B
Austenitic stainless steel																
304 SS	18 - 20	8 - 11	Bal.	0.08												
316 SS	16 - 18	11 - 14	Bal.	0.08 max	2.0 – 3.0											
Solid solution Ni-base austenitic alloy																
Alloy 600	15.5	76	8.0	0.08								0.5	0.2			
Alloy 690	30	60	9.5	0.03												
Precipitation hardened Ni-base alloy																
Alloy 625	21.5	61	2.5	0.05	9.0				3.6	0.2	0.2	0.2	0.2			
Alloy 718	19.0	52.5	18.5	0.04	3.0				5.1	0.5	0.9	0.2	0.2			
Ferritic/martensitic alloy																
HT-9	12	0.5	Bal.	0.20	1.0		0.5					0.6	0.4	0.25		
T-91	9		Bal.	0.10	1.0				0.08			0.45	0.4	0.20	0.05	
HCM12A	12		Bal.	0.11	0.4	1.0	2.0		0.05			0.6	0.1		0.06	0.003

Fabrication of the supercritical water loop system (SCWLS) was begun during the second quarter. Table 5 summarizes the activities on the construction of the system during the quarter. General and specific requirements for the main components including test vessel and load frame were determined and the main components were ordered. All the main components for the loop system were delivered in March and April. As of April, the load frame and vessel were installed and the pump and pre-heater were secured on the structure positioned next to the load frame. The main tubing for the loop was plumbed and electrical work including wiring and addition of three power sources was completed. All the sensors have been connected to the data acquisition system. Figures 10 through 14 show the overall view of the loop system and the main components. Some minor plumbing and wiring are required for the system to be functional. The fabrication is scheduled for completion by mid May. In May, the loop will be cleaned, and leakage and performance tests of the main components and of the system as a whole will be performed. At the same time, the data acquisition system will be tested.

Table 5. Summary of the activities on the construction of the supercritical water loop system during the second quarter.

Year	Month	Activity
2001	December	General and specific requirements for main components (test vessel, load frame, main pump, back pressure regulator, etc.) were determined.
2002	January	Main components were ordered.
	February	Minor components were designed. Fabrication of the loop system was started.
	March	Structures for components were designed and fabricated. U-bend sample holders were designed and ordered.
	April	Test vessel and load frame will be delivered and installed. The fabrication of the loop system will be completed.
	May	Cleaning of the loop and performance test of the system are scheduled to be completed.

At the meeting between MIT and the University of Michigan, the conditions for corrosion and SCC tests were also discussed. It was agreed that all of the same conditions would be used for both types of tests in the two laboratories. The conditions determined at the meeting are shown in Table 6. Initially, the parameters for Condition 1 will be followed. Variations in oxygen content and temperature are the only parameter variations decided at this time. Once the effects of these parameters are known, other test parameter variations will be considered.

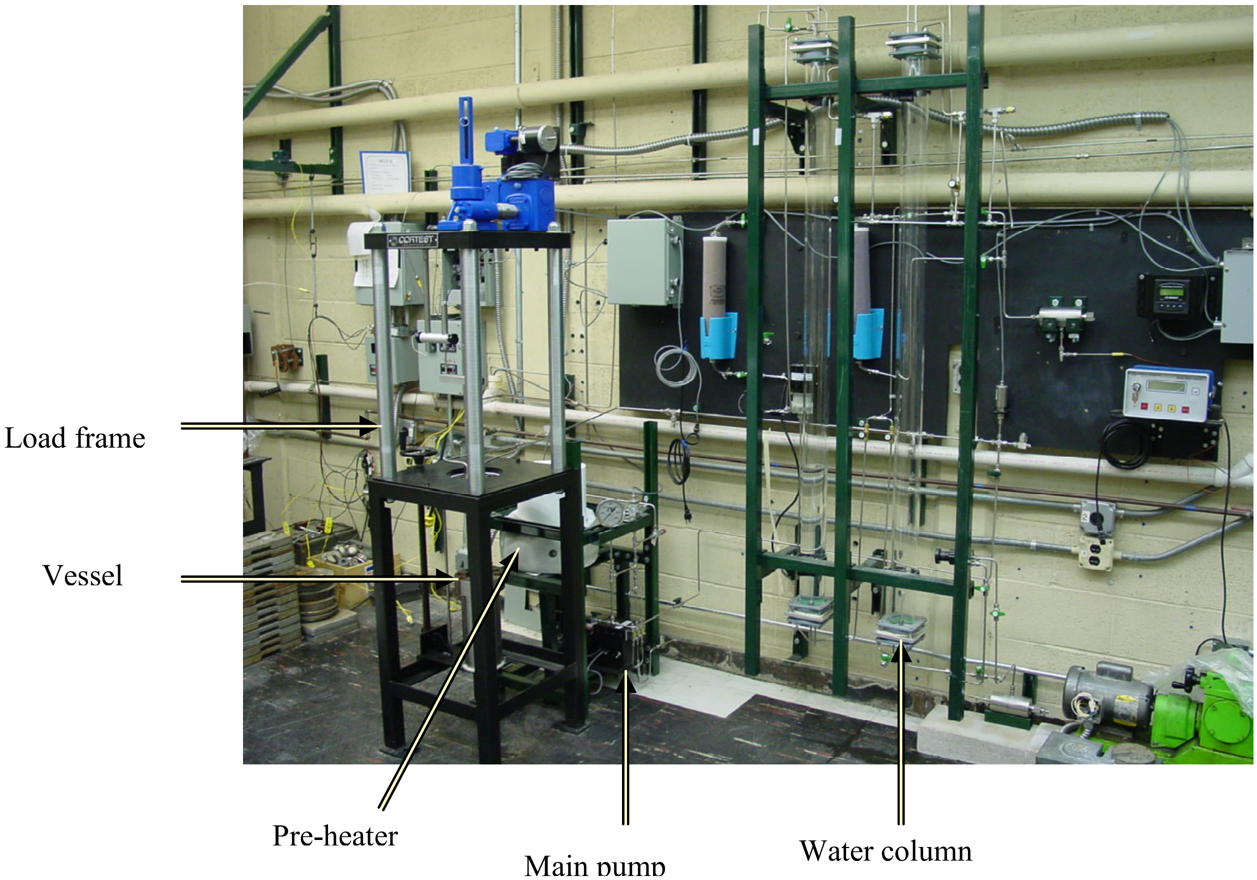

Figure 10. Overall view of the supercritical water loop system (SCWLS) in the High Temperature Corrosion Laboratory.

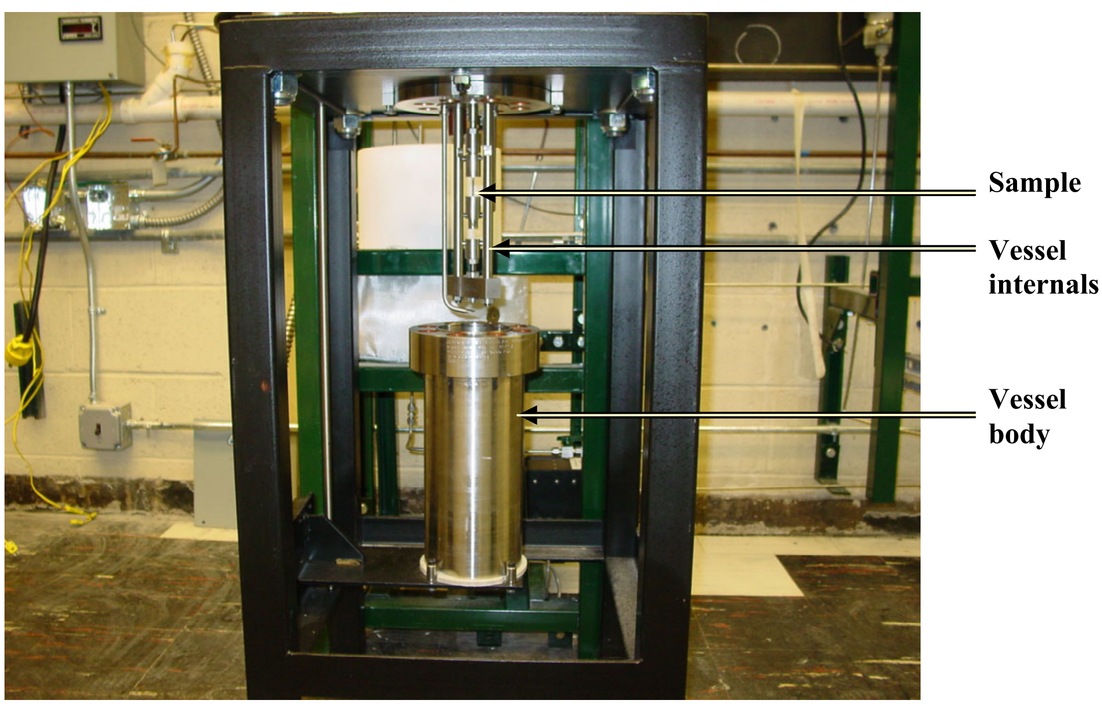

Figure 11. Supercritical water system: autoclave vessel and internals.

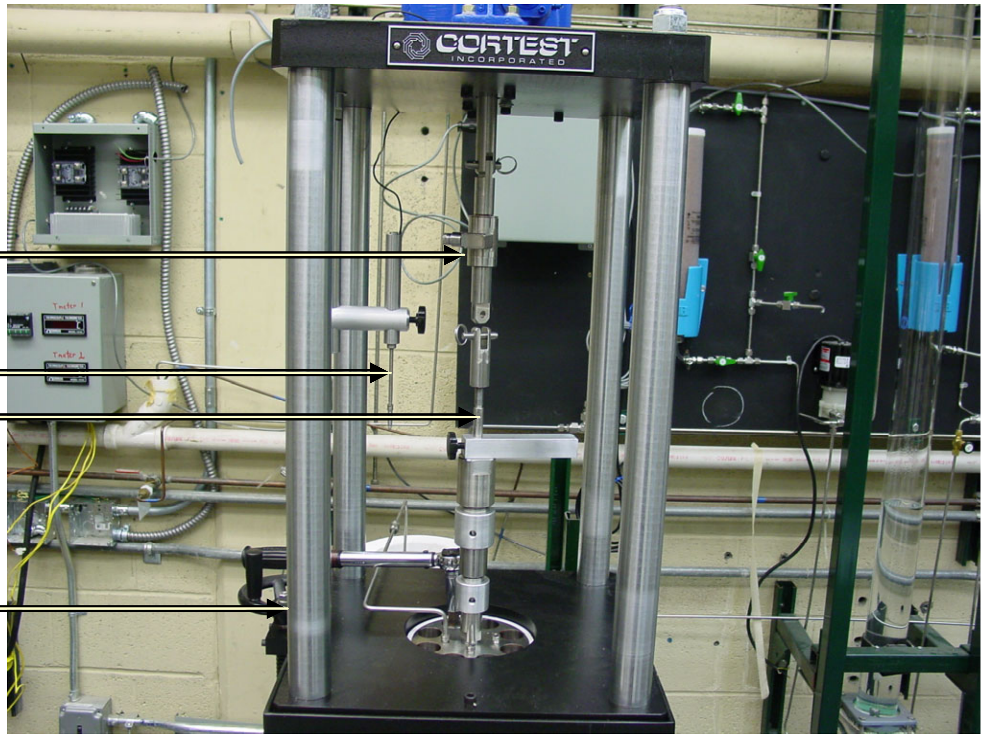

Figure 13. Supercritical water system: load frame and loading elements

Figure 14. Supercritical water system: main pump and pre-heater.

Table 6. Environmental conditions for SCW tests.

Condition	1	2	3	Remarks
Dissolved oxygen	Below 5 ppb	200 ppb	8 ppm air saturated at RT	i) Measured at RT and pressure below 20 bar. ii) Monitored continuously. iii) Measurable range = 1 ppb to 20 ppm
Temperature	Below 374°C	400°C	550°C	i) Monitored continuously. ii) Controlled within ± 1°C. iii) Max. T = 550°C
Conductivity	<0.1μS/cm	High (*)	X	i) Measured at RT on the inlet and out lines. ii) Monitored continuously. iii) Range: 0.55 μS/cm (18 MΩ cm - pure water) to 100 μS/cm (0.01 MΩ cm) with ± 2% reading error. (*) Depends on the concentration of ammonia added to adjust pH value.
Flow rate	10 ml/min	X	X	i) Between 1 and 100 ml/min at RT
pH	Neutral (7 at RT)	Caustic (9 - 9.5 at RT)	X	i) Measured at RT using periodically sampled water from the loop inlet.
Chemicals	No addition (pure water)	Ammonia (**) (NH_3)	X	(**) To adjust pH.

Task 3. Plant Engineering and Reactor Safety Analysis

Task 3.1 Conceptual Design of the Reactor Coolant System

Luca Oriani, Mario Carelli, Dmitry Paramonov, and Lawrence Conway

3.1.1. General Design Goals.

The Generation IV plants will have to not only represent a significant improvement over current nuclear plant designs in capital and operating cost, safety and public acceptance, and sustainability, but will also have to be competitive with other existing or developing power generation sources. The current capital cost goal of about $1000-1200/KWe for the near term deployment plants might not be sufficient in the longer time frame, and more aggressive targets of $700-800/KWe might have to be pursued. Given the long range R&D effort needed for development of SCWRs and the high expectations in reducing the capital costs, the proposed approach for this program is to try stretching the system performance to the largest possible extent.

The Japanese SCWR studies (Oka et al. 1993, Oka and Koshizuka 2000 and 2001) and the CANDU-MARK-1 (Spinks et al. 2002) program are focusing on supercritical designs that require a limited increase in temperatures above current reactor parameters. For the Mark-1, a core outlet temperature of 420°C is proposed and for the Japanese design of the SCWR, a core temperature of about 397°C has been proposed (a SCLWR-H with higher temperatures is also being studied). This is to allow the use of stainless steel cladding.

This development stage should be skipped and our goal should be to use coolant with the same temperatures as in fossil-fuel supercritical plants. Fossil-fuel supercritical plants have been built for the last 40 years to operate with steam pressure of about 3500psia (~24.1MPa) and temperatures of about 1050°F (565°C). Newer fossil-fired plant designs are being developed with even higher temperatures. With these higher temperatures, the efficiency of a nuclear plant would be around 45% (with a direct cycle, probably around 44-44.5% with indirect cycle: the difference in efficiency is small due to a saturation-like effect of the efficiency in the high temperatures range). Similar core outlet conditions are currently being considered also for the European High Performance Light Water Reactor, HPLWR, (Heusner et al. 2000) and for the Japanese SCLWR-H (Oka and Koshizuka 2000 and 2001).

The ability to exploit the high temperatures of fossil fuel supercritical plants will be probably dictated by finding new materials for the fuel cladding. Given the relatively low impact of the fuel cost to the overall cost of electrical generation in a nuclear plant, the high cost of present day Zircaloy cladding, and the very poor mechanical characteristics of zirconium, new cladding materials are not only needed, but probably can be cost effective. Different options are currently being considered for the cladding material that should allow cladding temperatures in normal operation up to 620°C / 1148°F.

In conclusion, our preliminary design will focus on a very high temperature reactor (*proposed design point: core outlet pressure at 25 MPa / 3625psia, core outlet temperature 560°C / 1040°F*). The thermal-Hydraulic and material performances will be evaluated to confirm this design point or eventually modify it.

3.1.2. Power Cycle: Direct Vs. Indirect Cycle and Other Considerations

The first critical choice that has to be made, and that will impact every features of the design (performance, safety, plant layout), is the adoption of a direct (BWR-like) or indirect (PWR-like) plant cycle.

Direct Cycle

The power cycle that has been previously proposed is a supercritical water, thermal-spectrum reactor with a direct cycle (SCWR-D, Super Critical Water Reactor, Direct cycle).

The **General Layout** would be very similar to a BWR, with the coolant (supercritical water) being heated in the reactor and then sent directly to the turbine.

The **Reactor Pressure Vessel** in this case would be more similar to a PWR than a BWR vessel, given the elimination of steam separators, driers, and re-circulation loops. Like a conventional BWR, the SCWR-D will be a once-through system. The Reactor Vessel should be extremely compact due to the very high power density (*future task: preliminary vessel sizing and drawings will be based on common PWR experience and on the core size*).

The **SCWR-D Balance of Plant** will rely heavily on the experience of fossil fuel plants so that no activity concerning the definition of the balance of plant is terribly urgent *(future task: we will develop the balance of plant model of the plant using appropriate tools, GE GateCycle?, to precisely evaluate the plant efficiency and for preliminary cost estimates)*. The balance of plant design would be similar to the BWR balance of plant design.

The system layout would be similar to the various SCLWR concepts proposed by Japanese researchers.

Indirect Cycle Layout.

Another solution that should not be discarded a-priori is that of a PWR-like indirect cycle. This solution presents some advantages and disadvantages that may not be immediately evident.

At a first glance, the strongest point against an indirect cycle is that of complexity and primary system cost, given by the need for steam generators and reactor coolant pumps. A loop, PWR-like, design will have to be similar to a B&W type PWR that makes use of once-through steam generators rather then the typical Westinghouse recirculation steam generators. Such a system was proposed by Westinghouse for the Supercritical, Closed-cycle Pressurized Water Reactor (SC-PWR) (WCAP 3374-8).

Other possible closed-cycle designs include a full natural circulation system which would require no reactor coolant pumps or an integral primary system type reactor, IRIS-like (Carelli et al. 2002, Collado et al. 2002) system. An integral primary system design would require a significant increase in the vessel size due to the need for locating the steam generators inside the reactor vessel and for mounting the eventual pumps on or even inside the reactor pressure vessel. Based on existing experience, an integral SCWR (ISCWR) solution would probably be better suited to small reactors with thermal powers up to about 1000 MWt (the maximum power dictated by concerns regarding an excessive RPV size/weight). The SC-PWR would instead be ideally applied to a high power plant, with thermal powers in the range of 3500-4000 MWt.

The balance of plant would not be significantly different from that of the direct cycle plant, except for the significant simplifications in design, control and maintenance of the balance of plant due to the presence of a non-radioactive fluid in the secondary system.

The sketches in Figure 15 below show the layout of the three considered solutions: Direct Cycle (15.a), Indirect Cycle in loop configuration (15.c) and Indirect Cycle in Integral Configuration (15.b)

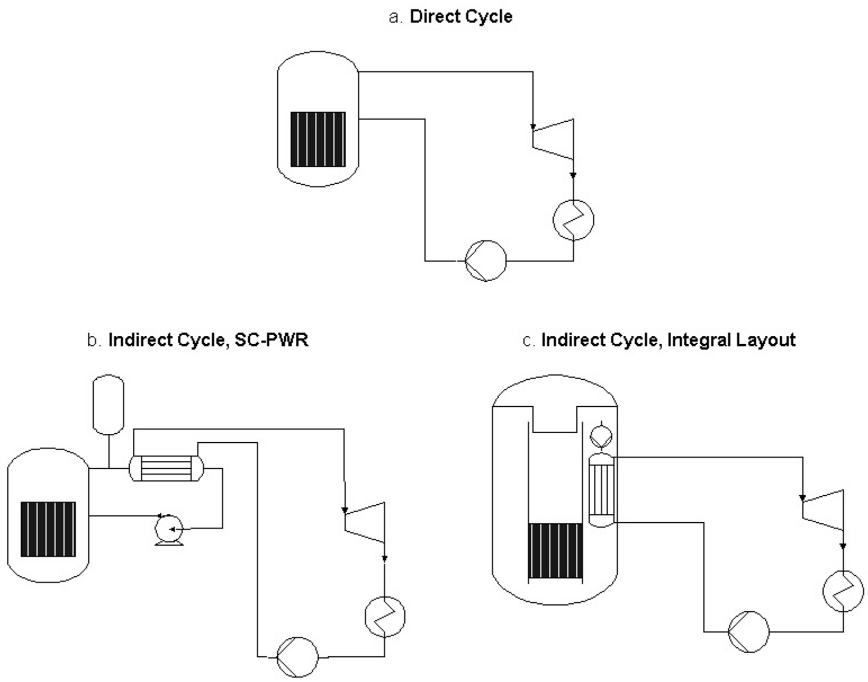

Figure 15. Different plant concepts for the Supercritical Light Water Reactor

3.1.4. Direct Cycle vs. Indirect Cycle: a Critical Comparison

The comparison between direct and indirect cycles is examined below since it provides some insights.

Selecting an indirect cycle solution would be equivalent to designing a BWR at the same temperatures and pressures of today's plants, and then making it an indirect cycle, PWR-like plant. Since the fluid can be directly sent to the turbine, why insert an intermediate stage? The answer is not so obvious as it might seem.

First of all, the loss in efficiency due to the adoption of an indirect cycle is not as high as could be expected because of the very high coolant temperature in the SCWRs. If we assume a core outlet temperature of 560°C for the supercritical plant, and that the steam generator outlet temperature will be lower by 50°C (a very conservative assumption: for IRIS this difference is about 12°C, and a supercritical steam generator will be much more compact and effective in exchanging power) the turbine inlet temperature for the direct cycle would be 560°C, and 510°C for the indirect cycle.

According to (Oka et al. 2002), the efficiency at 560°C is about 45%, but it is still ~44% at 500°C. Since a 60°C ΔT between primary and secondary is very conservative, the difference in efficiency between the two concepts will be small.

The difference between the two solutions will thus mainly be the cost of the steam generator and the increase in the cost of the vessel. Very roughly, for a 1000MWt reactor if we assume IRIS-like components, the increase in capital cost of the primary system can be anywhere between $50 and 100 million[1] (considering the ISCWR solution). If only this simple economic analysis was considered, the advantage of the direct cycle would appear as evident.

However, a very strong point can be made to show the potential equivalence of the direct and indirect cycle solutions based on a comparison with BWRs and PWRs. If we refer to Generation II Plants, it can be found that the actual efficiency of the two systems is very similar: the efficiency a BWR/6 is calculated as 32.9%, and the efficiency of a typical Westinghouse 4-Loop Plant (Sequoyah) is 33.7%. Looking at two Generation III concepts, efficiency for the GE-SBWR and GE-ESBWR is estimated as 33.5-34.5%, while the AP-600/AP-1000 efficiency is given as ~32 to 33.5%. The differences in efficiency between the two reactor concepts are therefore small, and can mostly be attributed to specific design differences rather than to intrinsic features of the direct vs. indirect concepts. Given this slight difference in efficiency, it can be assumed that the survival of both reactor concepts at the present time is an indication of a comparable Cost of Generation.[2]: BWRs and PWRs should have (and actually do have) more or less the same capital cost (or at least the differences should not be decisive).

This probably indicates that the increase in the primary system capital cost of a PWR (loop piping and steam generators) is offset by the complications in the RPV and in the turbine-generator system due to the presence of a radioactive fluid in the turbine and to several complications due to features of BWR steam cycle. Based on this preliminary consideration, it is difficult to make a definitive decision regarding the adoption of a direct or an indirect cycle, especially considering some advantages of the SCWR over a BWR in direct cycle applications (elimination of whole components of the RPV). The previous considerations would lead us to believe that, due to the similar efficiencies of the two concepts, the best estimate at this time would be for a comparable cost for the two options.

Moving from general considerations to specific issues, an indirect cycle application would have some advantages and some disadvantages. The disadvantages are listed first:

- **Pumps:** For a forced convection primary system, the main coolant pumps must be designed to operate at high pressure and temperature, well beyond the range of present day application. The most likely solution would be to rely on canned motor pumps that are positioned with the impeller up and the motor down. This would mean that the pumps would fill with cooled sub-cooled water and operate just as they would normally.

- **Steam Generators:** Steam generators for SCW applications should be relatively compact due to the high heat transfer coefficients. The lack of a phase transition should also help limiting inefficiencies in the steam generator design and thus achieving more effective and compact steam generators. Materials problems for the steam generator tubes will have to be carefully addressed. An interesting feature of such steam generators is that the primary and secondary pressure would be very similar in operation, but the steam generator tubes must be designed for full primary

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[1] Assuming a cost of vessel of ~$25 million for the direct cycle solution and a two to three fold increase for the indirect cycle, and a cost of the steam generators and eventual pumps of $25 (for Full Natural Circulation and no pumps, and our best guess at the steam generator cost) to $50 million. The cost of a supercritical plant to be competitive can be estimated at below $1000/Kwe, such that the total capital cost should be $450 million (2001 USD).

[2] Actually, this assumption is only partially justified. Industrial inertia would lead to the maintenance of both concepts even for significant differences in capital cost.

pressure. This means that the steam generator tubes will have very low stress in operation and thus failure of the steam generator tubes would be highly improbable.

- **Pressure Control:** A closed system will require the presence of a pressurizer of some sort, and given the fluid supercritical conditions a conventional steam-water pressurizer is not applicable to this case. The pressurizer shall have to be a gas-"water" system. Possible candidates as control gas are helium (proposed by Westinghouse for the SC-PWR), nitrogen (presently considered for Light Water Reactor applications), and argon. A gas pressurizer has significant issues, including solubility concerns (solubility is a function of density and this might lead to gas releases in the system), control during depressurization transients, and design of the gas injection/removal circuits. Another solution can be considered: the whole reactor coolant system can act as a pressurizer; i.e., a temperature increase decreases the density and raises the pressure, pressure is decreased if necessary by removing fluid to a small tank where the fluid is condensed and stored; pressure is then increased by adding water which is turned into supercritical fluid.

The advantages include:

- **Higher Core Inlet Temperatures**: The adoption of a closed system allows more freedom in the selection of core temperatures. One interesting solution would be to design the system so that both the inlet and the outlet temperatures are above the pseudo critical temperature. This will allow operating with a fluid having more homogeneous properties in the core and thus simplifying the core design and control. Moreover, a very low density of the fluid can be obtained in the entire core, thus increasing the interest and feasibility of a fast spectrum SCWR (fluid densities of about 100 kg/m^3 can be achieved). This will also significantly increase the flow velocity in the core region. Otherwise a complicated water tube design (2-way flow up and down and back up flow pattern) or a design where core flow is reversed, may be required to even out the moderator density seen by the core. On the other hand, designing for a large change of density, similar to the one used for the direct cycle solution, will probably allow for a natural circulation reactor coolant system. All in all, the adoption of a closed cycle gives more freedom in the definition of the optimal neutronic design of the system.

- **Natural circulation:** As compared to current BWRs, the direct cycle SCLWR design simplifies the vessel, but this eliminates any natural circulation recirculation path (the new BWR designs such as the SBWR and ESBWR have high levels of natural circulation). This means that following an accident, the feedwater must be maintained, and the outlet nozzle must be kept open to release energy. In comparison, an indirect cycle SCWR can use cooling via the steam generators so that the primary fluid can be isolated.

3.1.5. Conclusions And Future Effort Planning

Based on these preliminary considerations, three different options will be considered for further investigation:

1. Direct Cycle, with a thermal power in the range of 3000-4000 MWth. A different safety approach than the Japanese designs will be considered, both active and passive, taking as reference the GE ESBWR and the Advanced BWR.
2. Indirect Cycle Loop, with a thermal power in the range of 3000-4000 MWth, based on the Westinghouse AP600/AP1000 designs and experience.
3. Indirect Cycle Integral Primary System Reactor, with a thermal power around 1000 MWth, based on IRIS/SIR concepts.

Also, the above mentioned issues concerning some components of an indirect cycle system (steam generators, MCP, pressure control) will be assessed to verify the feasibility of the indirect cycle systems. A thermal reactor solution will be considered for all three approaches, with a core inlet temperature of ~280°C (based on typical values used for other supercritical concepts and on the fossil-fired plant typical regeneration rates).

For the direct cycle, the possibility of operating in the supercritical region (core inlet temperature above the critical temperature) will be explored as a possibility of achieving a fast reactor (since the fluid density will be very low in the whole core). A core inlet temperature between 400-450°C will be considered. This solution might also have some interest for a thermal core design, since it will lead to more uniform moderator densities in the core: the feasibility and eventual interest of this approach will also be evaluated.

Task 3.2 Definition of the Thermal/Mechanical Design Limits

3.2.1. Key Design Criteria: Light Water Reactors

As discussed in Task 3.1 above, the proposed preliminary point design for core outlet temperature and pressure are:

- Core Outlet Temperature: 560°C / 1040°F
- System Pressure: 25 MPa / 3625 psia

The definition of other thermal-hydraulic design parameters will require first the definition of a set of proper design basis criteria. In PWRs the commonly applied core design bases are:

- Maximum cladding surface temperature at nominal conditions (limit of 660°f for LWRs).
- Maximum fuel centerline temperature for nominal fuel rod dimension at 100% Power (typical limit of 3250°F).
- Minimum departure from nucleate boiling ratio (DNBR) at nominal operating conditions (depending on the system designed, usually the minimum DNBR has to be 1.7 to 2.5)

These criteria are then used in the definition of an acceptable core configuration (core geometry, flow rates, power densities, etc.).

DNB is also used as the main safety parameter for the core design. The primary operating protection point of a PWR reactor protection system is defined based on the "Core Limits". The core limits represent the loci of points of the thermal power, system pressure, and inlet temperature, at various pressures, which satisfy the following criteria:

1. The minimum DNBR is not less than the Safety Analyses Limit DNBR (SAL-DNBR) (e.g. a minimum DNBR of 1.3 is typically considered with several different design DNB correlations).
2. The hot channel exit quality is not greater than the upper limit of the quality range for the applicable DNB correlation.
3. The vessel outlet temperature is lower than the saturation temperature.

All of the above mentioned criteria clearly do not apply to supercritical reactors, so that a new design basis must be defined and this requires a careful consideration of the rationale behind the PWR design criteria. If the ANSI plant conditions summarized in Table 7 are considered, it appears evident that the minimum DNBR is used as an index of rod failure and thus minimum DNBRs are defined to satisfy the requirements for Condition I and II events. This is possible due to the significant heat transfer degradation that takes place once the crisis in heat transfer (critical heat flux) is reached, thus virtually guaranteeing that a rod in DNB will be damaged due to excessive cladding temperatures. Minimum DNBR thus becomes the limiting parameter in core design, but simply because it is the best index of fuel cladding damage.

Table 7. ANSI Plant Conditions

ANSI plant conditions.	Description	Effect on the Plant
Condition I : Normal Operation	Conditions accommodated with margins between plant parameters and values of parameters that would require automatic/manual protective actions.	No fuel damage expected (minimum DNBR within 95/95 rule[3]). Limited fuel damage within capability of plant cleanup system. Plant capable of operation after limited corrective actions.
Condition II: Incident of Moderate Frequency	Conditions of moderate frequency that can be accommodated, at worst, with reactor shutdown.	Same as Condition I.
Condition III: Infrequent Faults	Conditions of low frequency that will not, by themselves, generate a Condition IV fault.	Small fraction of fuel rods can be damaged. Immediate resumption of operation may be precluded. Release of radioactive materials should not be sufficient to interrupt or restrict public use of those areas beyond the exclusion radius.
Condition IV: Limiting Faults	Faults that are not expected to happen during the life of the plant but are defined as limiting faults against which the system must be designed.	Must not cause a release of radioactive material that results in an undue risk to public health. Resumption of operation may be precluded.

A deterioration phenomena somewhat similar to DNB has been observed also for supercritical water in transitions from below the pseudo-critical temperature to above, for very low flow velocities at high heat fluxes (Deterioration Heat Flux, DHF). A correlation has also been proposed to estimate this deterioration heat flux and thus a criterion similar to the minimum DNBR, the minimum DHFR has been proposed for studies involving water in supercritical conditions. However, DHF is a much milder phenomena then DNB: the differences in heat transfer coefficients in deteriorated flux conditions are only ½ to 1/3 the pre-deterioration region, and DHF only exists in the region around the pseudo-critical temperature. Moreover, if the high-proposed temperatures are considered, the minimum value of the heat transfer coefficient will not be in the DHF region, but at the higher temperatures. Finally, an Enhanced Heat Transfer (EHF) has been also observed in the same temperature region of the DHF phenomena for high flow velocities.

[3] The 95/95 rule is the criteria commonly used to verify DNB margins with a statistical approach. 95/95 means that there is a 95 % probability with a 95% confidence that DNB will not occur for the most limiting rod. Historically, this criterion has been conservatively met by designing for a minimum DNBR of 1.3 with Westinghouse correlations (W-3).

The DHF is therefore not going to be an index capable of replacing the DNBR as a design parameter for supercritical reactors. In supercritical conditions, the cladding temperature is probably the best parameter to be used (in this a supercritical reactor is probably more similar to a liquid metal reactor). Core limits for a SCWR should, therefore, be defined in terms of acceptable cladding temperature values, rather then in terms of acceptable DNBRs. However, the procedure for defining these design criteria is not as straightforward as it may appear. Given the high fuel temperatures, probably two different criteria will have to be defined:

1. **Steady state maximum cladding temperature**: this should be defined as the maximum temperature allowable in nominal conditions to guarantee the optimal performance of the fuel rod, thus limiting corrosion rates to acceptable values. A design parameter can be defined as the minimum NCTR (Nominal Cladding Temperature Ratio). This is the ratio between the maximum calculated design cladding temperature and the defined cladding limit temperature. This parameter will be more connected to maintaining acceptable fuel performance (i.e. corrosion rates) than to safety.

2. **Safety analyses maximum cladding temperature**: in safety analyses, probably higher values of the cladding temperature can be allowed (short periods of time, versus long periods of time considered in defining the minimum NCTR). The maximum allowable temperature should be defined under the usual NRC 95/95 rule. This means that the maximum allowable temperature will be that temperature that guarantees the integrity of the fuel rod with a 95% probability and a 95% confidence. How to define this value is probably not very simple. As for the minimum NCTR, an appropriate ratio can be defined as the minimum Safety Cladding Temperature Ration (SCTR) as the ratio between the maximum calculated transient cladding temperature and the defined cladding safety limit temperature.

The minimum NCTR and the minimum SCTR will become the key parameters in defining the core limits for the Supercritical plant. These two values will have to be defined by the groups involved in the selection of suitable cladding material.

One important point is that shifting the interest from the minimum DNBR to the cladding temperature will require the development of new supercritical water heat transfer correlations to remove unnecessary conservativness from the analysis: heat transfer correlations will in fact be used in the same way CHF correlations are used in present day LWR to define safety margins. These new correlations will have to be developed specifically for the core geometry that will be defined, and will have to include the effect of DHF and EHF to allow for a proper and complete analysis of the clad temperature.

In addition to the cladding temperature, the fuel centerline temperature will play an important role: given the higher temperatures of the SCWR, this can become an even more important parameter for safety analyses. The value of the maximum fuel temperature can be defined based on common PWR experience.

In conclusion, the following design/safety criteria can be considered for a preliminary design:

1. Minimum NCTR (Nominal Cladding Temperature Ratio). This will require the definition of a cladding surface temperature that will guarantee the proper operation of the plant, and that will guarantee the life of the cladding.
2. Minimum SCTR (Safety Cladding Temperature Ratio). This will require the definition of a temperature of the cladding that satisfies the 95/95 rule.
3. Maximum Fuel Temperature for Nominal Conditions: 1790°C (3250°F).
4. Maximum Fuel Temperature for Transient Analyses: 2600°C (4700°F).

The core limits for SCWRs will probably be defined as the loci of point of the thermal power, system pressure, and inlet temperature at various pressures that satisfy the following criteria:

1. The minimum NCTR (Nominal Cladding Temperature Ratio) is not less than the design limit minimum NCTR (DL-MNCTR) (a DL-MNCTR will have to be defined statically based on the heat transfer correlation uncertainties).
2. The hot channel exit enthalpy is not greater than the upper limit of the enthalpy range for the applicable heat transfer correlation.
3. The vessel outlet temperature is lower than the vessel design temperature.

References

- Briesmeister, J. F., 1997. MCNP - A General Monte Carlo N-Particle Transport Code (version 4b), Los Alamos National Laboratory, Los Alamos, NM.
- Carelli, M., et al., "Iris: Proceeding Towards The Preliminary Design", Proc. of 10th International Conference on Nuclear Engineering, ICONE10, Arlington, VA, April 14-18, 2002
- Collado, J., et al., "Pressure Vessel And Internals Of The International Reactor Innovative And Secure (iris)", Proc. of 10th International Conference on Nuclear Engineering, ICONE10, Arlington, VA, April 14-18, 2002
- Heusner, G., et al., "A European Development Program for a High Performance Light Water Reactor (HPLWR)", Proc. of SCR-2000, Tokyo, Nov. 6-8, 2000
- Oka, Y., et al, "Concept and Design of Supercritical-Pressure, Direct Cycle Light Water Reactor", Nuclear Technology, Vol. 103, pp 295-302, 1993
- Oka, Y., S Koshizuka, " Supercritical-Pressure, Once-through Cycle Light Water Cooled Reactor Concept", Journal of Nuclear Science and Technology, Vol. 38, No. 12, pp. 1081-1089, 2001
- Oka, Y., S Koshizuka, "Design Concept of Once-through Cycle Supercritical Pressure Light Water Cooled Reactor", Proc. of SCR-2000, Tokyo, Nov 6-8, 2000
- Oka, Y. et al., "Core and Plant Design of the Power Reactor Cooled and Moderated by Supercritical Light Water"
- Spinks, N., et al., "Thermo-Economic Assessment of Advanced, High-Temperature CANDU Reactors", Proc. of 10th International Conference on Nuclear Engineering, ICONE10, Arlington, VA, April 14-18, 2002
- WCAP-3374-8, "Supercritical Technology program- Final Report", May 1968

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

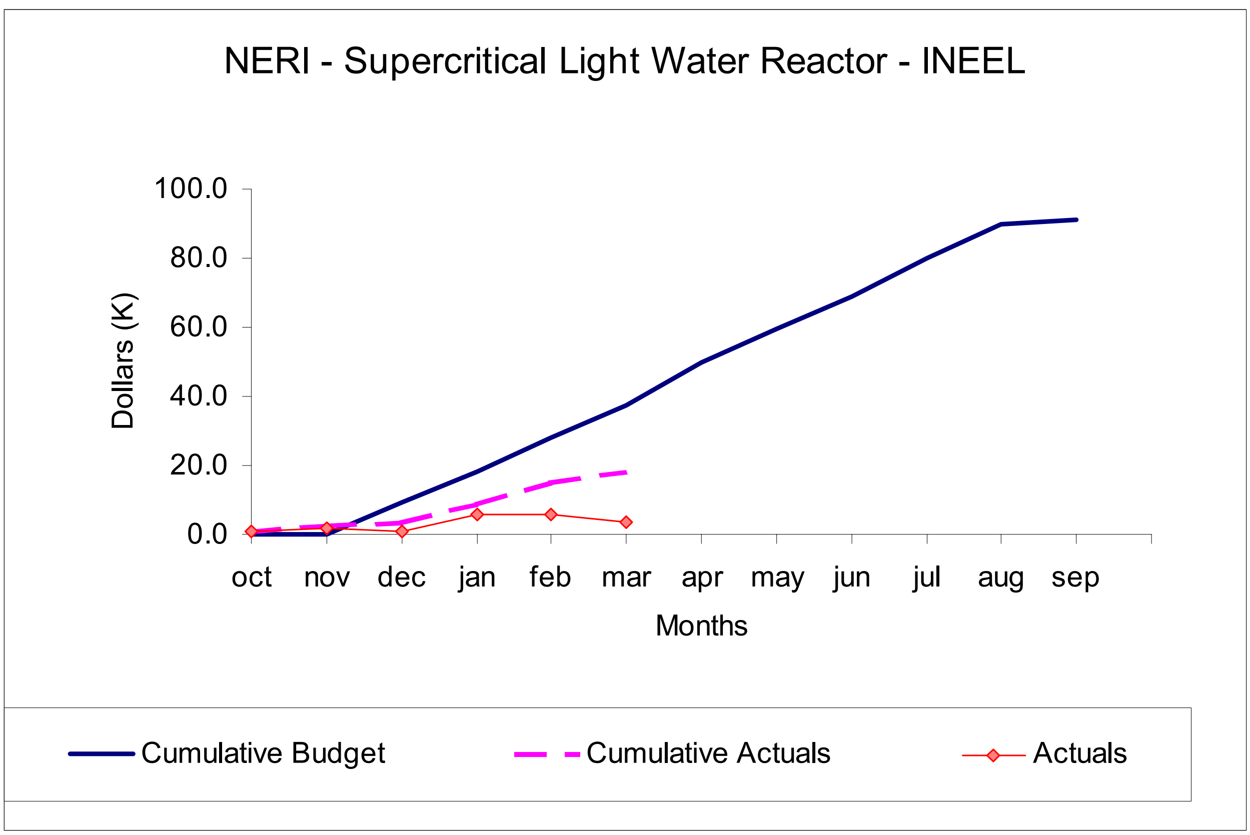
Summary Budget

Organization	Year 1 (Budget)	Year 1 Actuals, September 2001-March 2002	Year 1 Actuals, September 2001 –May 26, 2002
INEEL	107.7.0K	18.4K	52K
University of Michigan	142.7K	73.9K	101K
MIT	46.6K	24.5K	34K
Westinghouse	100K	0.0K*	20K
Total	397K	116.8K	207K

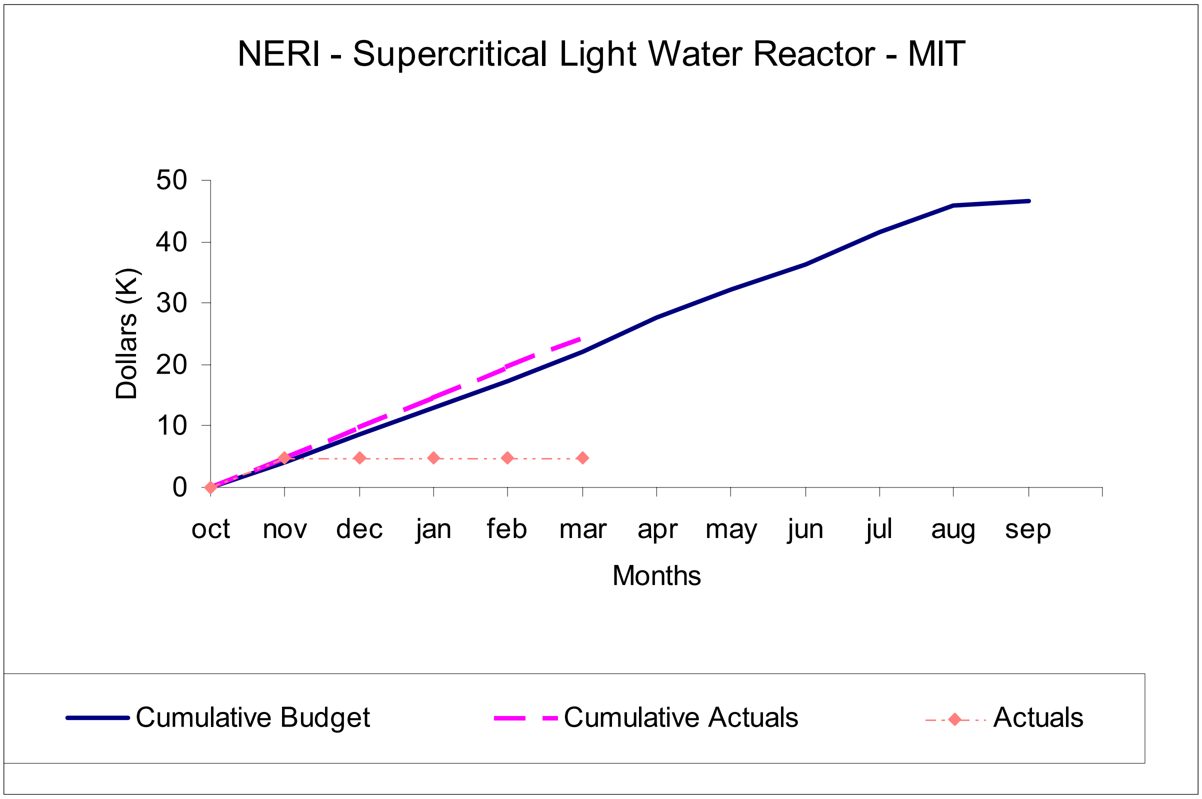
* Contract placed after March 2002.

Budget Plots

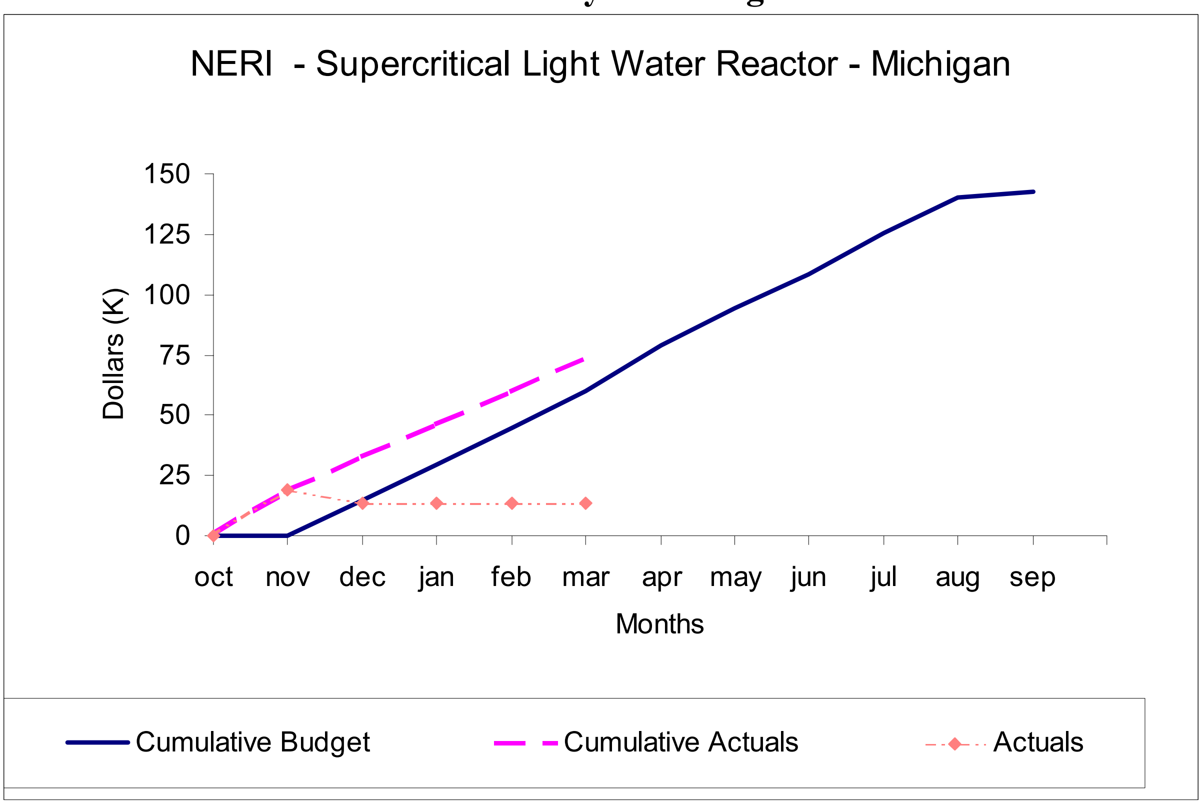
INEEL



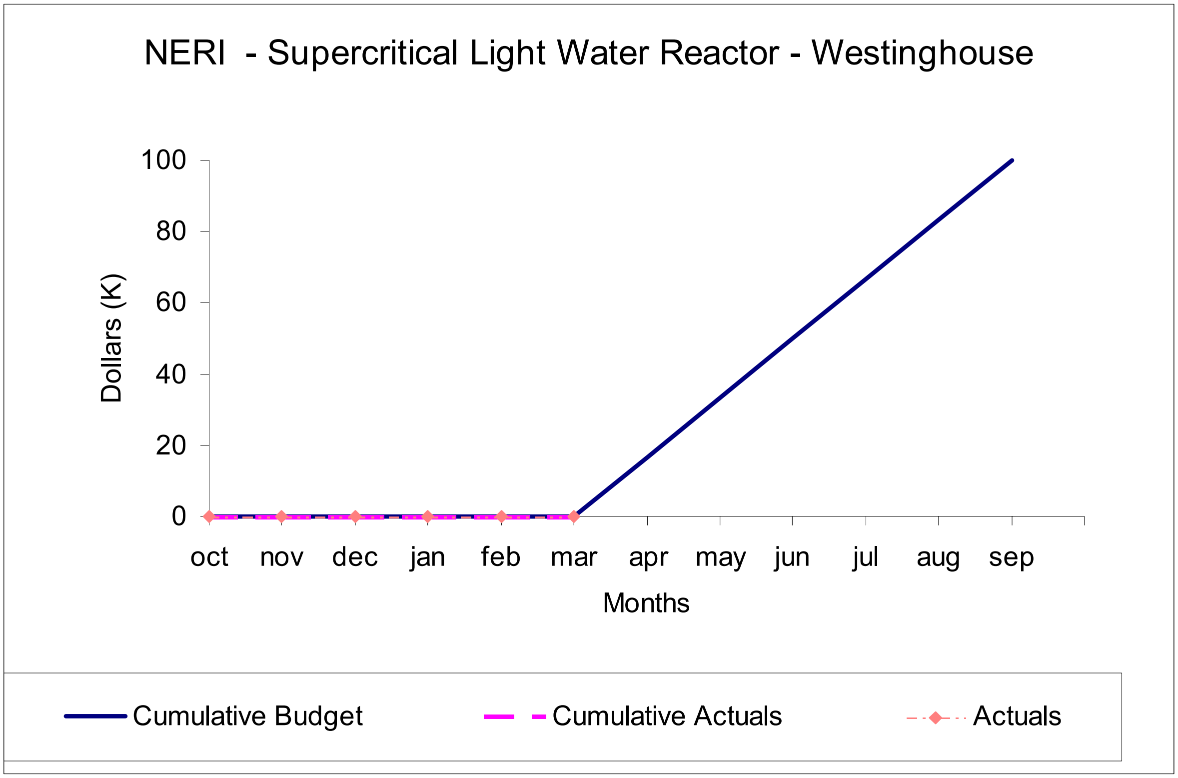
MIT



University of Michigan



Westinghouse Electric Co.



INEEL/EXT-02-00759

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 2 (January through March 2002)

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Principal Investigator: Professor Gary Was

Westinghouse Electric Company
Principal Investigators: Luca Oriani, Mario Carelli, Dmitry Paramonov, and Lawrence Conway