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LIQUID METAL FAST REACTOR TRANSIENT DESIGN

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Abstract

An examination has been made of how the currently available computing capabilities could be used to reduce Liquid Metal Fast Reactor design, manufacturing, and construction cost. While the examination focused on computer analyses some other promising means to reduce costs were also examined.

Introduction

A major problem with commercialization of Liquid Metal Cooled Fast Reactors has been high design and manufacturing costs. Operation with coolant and metal structures temperatures in the creep range, coupled with sodium's high thermal conductivity necessitate use of codes, standards, requirements, and design approaches that have been very costly.

While it has been accurately stated that the cost of high temperature design on FFTF and Clinch River was small compared to the cost of the components, this design cost was still very high by any measure. Liquid metal reactor components and systems designers must perform extensive strain limited structural analyses, address considerations associated with phenomena such as creep fatigue interactions, and provide for the design to accommodate relatively large movements and flexibility. An additional expense was associated with the effort related to code interpretations and code cases. Considerable effort was involved in the use of simplified methods (screening procedures) to pinpoint problem areas and minimize the more elaborate analyses of critical structural configurations required to show code compliance. This made it possible to resolve design and analysis problems, but restricted choices system and component design.

Much larger costs than the obvious costs directly associated with high temperature design were associated with the impact of this required methodology on the resulting designs and design requirements for systems and components.

Design transients and operating transients

The duty cycle (design) transients are events selected, by the designer, as representative of operating conditions that have been determined may occur during plant operation and that are sufficiently severe or frequent to be of possible significance to component behavior. These events are classified as normal events, upset events, emergency events, and faulted events. Attachment 1 provides a definition of these categories of events and a short discussion of the severity of damage and significance of the event categories.

These transients are not normally expected to represent actual plant operations, but are meant to be used for predictions of systems response to the events and for component stress analyses. The systems analyses include thermal/hydraulic analyses of the coolant, neutronic analyses of core behavior, coupled thermal/hydraulic-neutronic analyses to allow consideration of the effects of interactions, and analyses using coolant temperatures to predict metal temperatures, temperature rates of change and the like. In addition, it was necessary to build scale models on do testing (e.g., hydraulic, mixing, vibration) to obtain information for both direct use and for use in dimemsionless analyses. Simplified analyses were used to idealize complex geometries, loading histories, and material models. These screening analyses were used mainly to identify highly stressed and critical areas to guide design choices.

Attachment 2 provides a listing of the design transients for the Clinch River Breeder Reactor Plant. These events and frequencies provided are from Revision 118, the last Revision before the project was terminated. These design transient events and frequencies were used to order essentially all of the components for Clinch River. Most of the design work was completed. All major components were either delivered or well into manufacturing. The plant had essentially completed the licensing process, with all major issues having been resolved. Revision 118 is a mature listing of transients that had been used in the real world to design a loop type LMFR.

These demonstrably conservative transients were then grouped into umbrella transients and applied to various components and locations in components. This grouping of transients resulted in the frequencies of some relatively minor transients being summed with the most severe transient in the group. Thus the selected umbrella design transients are normally very conservative both as to severity and number.

The actual operational transients experienced should be considerably less severe than the design transients and have a significantly lower frequency of occurrence. The use of the actual operating transient history, the loading conditions accompanying these events, and the frequency can be used to reevaluate to design, for purposes such as life extension. These analyses should demonstrate that the achievable operating life of the plant is much greater than the originally specified design life.

Past practices

Experience showed that minor differences of opinion on analysis procedures concerning the ASME Code and code cases exaggerated difficulties and had considerable impact on increasing cost and restricting design options. The restrictions resulted from the inability to perform the large number of complex analyses required for many options due to costs, lack of adequate computing power, and related factors. The result was the need to select approaches that avoided, or minimized, uncertainties and to rely on the simplified, screening analyses, with the attendant restrictions in design options.

Differences in design practices have in the past handicapped developments that could have reduced costs. For example, some difficulties with past design practices are evident in examining application of the ASME Code and Code Case N-47 to pool type LMFRs, such as Super Phenix. The design of Super Phenix resulted in considerably thinner components. This required different buckling rules. Super Phenix design creep effects were negligible during normal operation, requiring different treatment from loop type reactors during transients into the creep range, such as resulted during emergency and faulted conditions. Because of these and other such problems the French developed a set of LMFR rules to address shortcomings of the ASME Code and apply available design and construction experience.

Needs to address problem areas in high temperature structural analyses were studied by the Working Group, Codes and Standards (WGCS) of the Commission of the European Communities in 1979. The WGCS examined and promoted efforts on benchmark calculations, constitutive equations, fracture mechanics, and seismic analyses.

Work to resolve these and other questions has been substantially reduced in recent years. Very few of these problems that rely on obtaining long term materials data have progressed. Some work on alternate materials to austenitic stainless steels has produced results.

Approach and limitations

An examination was made of the work that was done for Clinch River design using these transients. FFTF design work on reactor components was also considered. This effort focused on consideration of what would be done different at the present using currently available computer technology. The evaluation was qualitative, since there was no reasonable way to determine exact savings in design, analysis, and manufacturing costs.

There was one key technology, other than computer technology considered. This was the possible use of 9 Cr - 1 Mo, a ferritic/martensitic alloy. 9 Cr - 1 Mo promises to be an excellent substitute for austenitic stainless steels and 2-1/4 Cr 1 Mo steel. This material could be used in the entire system, thus eliminating dissimilar metal transition joints. It offers resistance to irradiation induced metal swelling and creep, helium embrittlement, and would allow higher design margins for ratchetting and creep-fatigue in steam generator applications.

In the past considerable effort has been devoted to finding and using "short cuts" to avoid the necessity of what were considered to be impracticably complex analyses.

The steps in the design process were examined. Areas where currently available computing power and analytic capabilities offered promise for cost reductions were identified. The potential reductions in manufacturing costs by increases in design flexibility were identified. Areas where no potential improvements could be identified at this time where identified.

Design areas and potential cost reductions

Systems design requires the interaction of a number of different disciplines with different interests. Examples are instrumentation and control, component design, licensing, and operation people. Design transient events must be realistic from the stand point of the instrumentation and control systems. They must be achievable by the component designers. They must be acceptable to the nuclear regulatory authority. And last, but not least they must be acceptable to the operations people. For this reason the operations people were involved from the start. Preparation of operating procedures was done in parallel with design analyses.

Due to the cost of the extensive analyses required, it has been the practice of designers to "lump" transients together as "umbrella" transients. Thus, instead of having a duty cycle requiring thousands of analyses of events that may occur only once or at most a few times, there is a duty cycle with a small number of events. The price paid is that the resulting "demonstratively conservative duty cycle results in the need for extremely expensive design solutions.

This duty cycle creates challenges that have been met by designers of liquid metal reactors that operate at high temperatures, such as the Fast Flux Test Facility and the Clinch River Breeder Reactor Plant. However, this was done at great cost for the analyses, extremely high manufacturing costs, and has created a disconnect between design transients and the real world.

Examination of the traditional design process, screening analyses, selection of the duty cycle, umbrella transients, unresolved differences in national codes and standards, inability to quantify the safety margins involved in use of codes and standards, and actions that overlay conservative actions on conservative actions shows that this process, while demonstrated to be highly effective, is also extremely expensive. The extreme conservatism involved in this design process is not visible, with the result that the public cannot see or understand the level of design conservatism and safety that results. Worst yet, many designers also do not understand. This results in apparently conservative actions that may do nothing to improve the design.

A new design approach should be possible based upon current computing and information technology. Such an approach could result in significant design simplifications and reduce the cost of manufacturing and construction.

It is now appears possible for designers of liquid metal fast reactors to achieve the same results at considerably lower cost. The historical practice of "umbrella" transients and the associated simplified duty cycle is no longer necessary. The current computing capability of easily affordable PCs and work stations now makes it possible to analyze the many transients that must be considered in the design and develop more economical structural and systems design solutions. Future Liquid Metal Reactors can take advantage of computer technology to significantly lower costs.

To determine the systems response to transients it is necessary to prepare a model of the entire reactor system. This can be done at several levels of complexity, with areas such as the core or the steam generators being represented in considerable detail, or relatively simple. The transient model can readily be a simulator. The use of "masks" to allow ease of modeling has been demonstrated in Nuclear Plant Analyzers developed for various nuclear power plants including VVERs in various countries. The use of the available spectrum of modeling complexity allows the systems designer to work with the various specialized disciplines to obtain a detailed understanding of the response of various systems to various design options. Areas of special concern can then be modeled in greater detail, or more detailed simulations run on the more promising design options.

It should not be necessary to develop, early on, a restrictive duty cycle to be applied as a design specification. The potential for ease of modeling can allow optimization of the transients that would be characteristic of various design approaches. Specific structural responses to these transients can be determined and changes made to mitigate problems.

This would entail a much more elaborate effort in the conceptual design phase that would reduce design uncertainties in the later phases.

Rather than work to achieve a simplified duty cycle and simplified umbrella transients, more exacting and realistic analyses could be performed. The currently available computing power allows these multiple analyses to be performed, facilitating optimization and removing the necessity for very conservative approaches to be taken to minimize and simplify calculations and analyses. At the same time the actual safety margins can be much better understood and more transparent, thus resolving one of the past difficulties between US and European design requirements.

By building on past operating and modeling experience it should be possible to use computer modeling and thus reduce the requirement for, or extent of, model or component testing.

The use of advanced computer modeling should allow reduced costs of fabrication, inspection and surveillance. In the longer term, it should be possible to develop uniform international code approaches for LMFR design that are less restrictive and allow improvements in plant availability and reliability.

There are some areas where it may not be currently practical to realize improvements using computer technology. There remains a need for improved materials data. However, it is possible that considerations of current computer analysis capabilities might result in changes in requirements for material properties information. For example, in some locations thermal striping causes a very large number of repetitive thermal transients on the metal surface being washed by sodium of fluctuating temperatures. It has been necessary to design for an infinite number of cycles. This requires the use of materials such as inconel, with associated high material and manufacturing cost.

An area where savings may be possible with improved materials data is on evaluation of welded joints. The material properties of the complete weld joint, as opposed to the weld metal alone, are needed. Ductility under multiaxial loading plays a key role in the structural adequacy of weldments subject to cyclic loading.

The US Liquid Metal Fast Reactor program compiled Nuclear Systems Materials Handbook. There is a vast amount of published information concerning structural design and evaluation of nuclear power plant components and high temperature structural design. Of late the quantity of such publication has greatly diminished. There is and will be, however, significant information coming from activities such as the Phenix Life Extension Project, from ongoing development activities, from applicable non nuclear work, and potentially from other activities, such as at Dounreay.

A related point is the need for an internationally available, and maintained, nuclear materials data handbook. The existence of the vast amount of potentially usable information, coupled with the extreme difficulty in searching for and locating this information is totally incompatible with current information technology. For example, while it is easy to find much information on many subjects on the Internet, it is virtually impossible to find any information on topics related to high temperature structural design and materials data. This data is published in had hoc reports, in a number of technical journals, and in various reports that are not generally available. As people who know where the information is leave the field these documents become increasingly hard to find and use. The result is that much valuable information, obtained at great expense, is being lost. Action to correct this situation would be of significant benefit.

There has been, and continues to be, significant High Temperature Structural Design technology developments for complex, critical non-nuclear structures that are subjected to elevated temperatures during normal operation. For example, such activities have been conducted in many nations to support aerospace development programs, such as those related to engines.

Because these developments are focused on intended applications in other fields and not problems of LMFR design there are significant differences in design lives, service conditions, materials, manufacturing practices, etc. The types of structures differ. The impact of these differences on such design information as constitutive models, material failure modes and models, and structural failure modes and consequences are sometimes difficult to assess. However, computer modeling, structural analyses methods, and analytic methods to understand materials behavior have advanced greatly in some of these non nuclear areas.

It is obvious that application of these developments in non nuclear areas is not a trivial undertaking. In spite of the obvious difficulties adapting these developments to LMFRs offers considerable promise and should be aggressively pursued. The reductions in the overall level of LMFR development activities, with the attendant reduction in work specifically directed at LMFRs makes such effort doubly attractive.

Attachment 1 **Definitions of categories of events**

Normal: Normal operation includes steady power operations and those departures from steady operation which are expected frequently or regularly in the course of plant operations, refueling, maintenance, or maneuvering of the plant. These events are to cause no damage. No damage is defined as those that:

1) result in no significant loss of effective fuel life;
2) are accommodated within the fuel and plant operating margins without requiring manual or automatic protective actions; and
3) result in no planned release of radioactivity.

Upset Any abnormal incident not causing a forced outage or causing a forced outage for which corrective action does not include any repair or mechanical damage. These off-normal conditions can cause anticipated conditions which individually may be expected to occur once or more during the plant lifetime. These operational incidents are occurrences that:

1) result in no reduction in effective fuel lifetime below the design values;
2) can be accommodated with, at most, a reactor trip that assures the plant will be capable of returning to operation after corrective action to clear the trip cause; and/or
3) Result in plant radioactivity releases that may approach the 10CFR20- guidelines.

Emergency Infrequent incidents requiring shutdown for correction of the condition or repair of damage in the system. There is no loss of structural integrity. These include unlikely off-normal conditions which individually are not expected to occur during the plant lifetime. However, when integrated over all plant components, events in this category may be expected to occur a number of times. These may result in minor incidents, that is an occurrence which results in:

1) a general reduction in the fuel burnup capability, and at most, a small fraction of fuel rod cladding failures:
2) sufficient plant of fuel rod damage that could preclude resumption of operation for a considerable period of time; and/or
3) plant radioactive releases that may exceed 10CFR20 guidelines, but does not result in interruption or restrictions of public use of areas beyond the exclusion boundary.

Faulted Postulated event and consequences where integrity and operability may be impaired to the extent that considerations of public health and safety are involved. These are off-normal conditions of such extremely low probability that no events in this category are expected to occur during the plant lifetime, but which represent extreme or limiting cases of failures which are identified as design bases. These are major incidents which can result in:

1) substantial fuel and/or cladding melting or distortion in individual fuel rods, but the configuration remains coolable;

2) plant damage that may preclude resumption of plant operations, but that will not cause loss of safety function necessary to cope with the occurrence; and/or

3) radioactivity release that may exceed the 10CFR 20 guidelines, but are well within the 10CFR100 guidelines.

Attachment 2

Clinch River Breeder Reactor Plant
Design Transient Events and Frequencies
(From Overall Plant Design Description, OPPD 10, Revision 118)

The **Event** is a short description of events to be considered in the plant structural design (see Attachment 1).

The **Frequencies** are the maximum number of occurrences of each event expected during the life of the plant. *These frequencies are used as the basis for plant structural design.*

A. Normal Events and *Frequencies*

N-1 Dry system heat up and cool down, sodium fill and drain loop for an entire system
5 total system + 8 per loop + an additional 17 per intermediate loop, exclusive of the Intermediate Heat Exchanger.

N-2a	Startup from refueling	*140*
N-2b	Startup from hot standby	*700*
N-3a	Shutdown to refueling	*60*
N-3b	Shutdown to hot standby	*210*
N-4a	Loading and unloading	*9300 (each)*
N-4b	Load fluctuations	*46~500 (each, up and down)*
N-5	Step load changes of +10% of full load	*750 (each)*
N-6	Steady state temperature fluctuations	*30×10^6*
N-7	Steady state flow induced vibrations	*10^{10} (Sodium)*

B. Upset Events and *Frequencies*

Note: The total frequency for U-1 events is associated with normal decay heat so as to balance the trips associated with partial decay heat for events U-2 through U-23.

U-1a	Reactor trip from full power with normal decay heat	*180*
U-1b	Reactor trip from full power with minimum decay heat	*0*
U-1c	Reactor trip from partial power with minimum decay heat	*0*
U-2a	Uncontrolled rod insertion	*10*
U-2b	Uncontrolled rod withdrawal from 100% power	*10*
U-2c	Uncontrolled rod withdrawal from startup with automatic trip	*17*
U-2d	Uncontrolled rod withdrawal from startup to trip point with delayed manual trip	*3*
U-2e	Plant loading at max. rod withdrawal rate	*10*
U-2f	Reactor startup with excessive step power change	*50*

Note: These events are part of the startups specified for event N-2b and should not be added as separate startups.

U-3a	Partial loss of primary pump	*2 per loop*
U-3b	Loss of power to one primary pump	*5 per loop*
U-4a	Partial loss of one intermediate pump	*2 per loop*
U-4b	Loss of power to one intermediate pump	*5 per loop*
U-5a	Loss of AC power to one feedwater pump motor	*10*
U-5b	Loss of feedwater flow to all steam generators	*5*
U-6	(Deleted)	
U-7a	Primary pump speed increase	*5*
U-7b	Intermediate pump speed increase	*5*
U-8	Primary pump pony motor failure	*5 per pump*
U-9	Intermediate pump pony motor failure	*5 per pump*
U-l0a	Evaporator module inlet isolation valve closure	*4 per loop*
U-10b	Superheater module inlet isolation valve closure	*2 per loop*
U-l0c	(Deleted)	
U-l0d	Superheater module outlet isolation valve closure	*2 per loop*
U-11a	Water side isolation and dump of both evaporators and the superheater	*6 per loop*
U-11b	Water side isolation and dump of evaporator module	*6 per loop*
U-11c	Steam side isolation and dump of superheater	*3 per loop*
U-12	Loss of feedwater flow to one steam generator loop	*3 per loop*
U-13	Feedwater throttle valve failed open	*6 per loop*
U-14	Loss of one recirculation pump	*8 per loop*
U-15a	Turbine trip (without reactor trip)	*50*
U-15b	Turbine trip with reactor trip (loss of main condenser or similar problem)	*10*
U-16	(Deleted)	
U-17	(Deleted)	
U-18	Loss of all offsite power	*16*
U-19	Plant shutdown in response to small sodium-steam/water leak indications	*3 per loop*
U-19a	(Deleted)	
U-19b	(Deleted)	
U-19c	(Deleted)	
U-20a	Inadvertent opening of one turbine bypass valve	*5*
U-20b	Turbine bypass valve fails open following reactor trip	*5*
U-21a	Inadvertent opening of evaporator outlet safety power relief valves	*5 per loop*
U-21b	Inadvertent opening superheater outlet safety/ power relief valves	*3 per loop*

U-22 Inadvertent opening of SGAHRS steam drum vent valve *3 per loop*

U-23 Inadvertent opening of evaporator inlet dump valve *3 per loop*

U-24 Reactor trip with failure of one PACC to perform *10*
Note: These events are part of the reactor trips for event U-la and should not be added as separate trips.

C. **Emergency Events**

The frequencies for these events are that each component must accommodate 5 occurrences of the most severe emergency transient for that component (one every 6 years) plus two consecutive occurrences of the most severe event (or consecutive occurrences of 2 unlike events if the unlike events provide a more severe effect than consecutive occurrences of the most severe event). However, if event E-15 is the most severe condition for a component] *it shall be evaluated for a frequency of 2 for that component in addition to the 7 occurrences of the next most severe transient.*

E-l Primary pump mechanical failure

E-2 Intermediate pump mechanical failure

E-3a (Deleted)
E-3b (Deleted)
E-4a (Deleted)
E-4b (Deleted)
E-4c (Deleted)
E-4d (Deleted)

E-5 Loss of primary pump pony motor with failure of the check valve to shut

E-6 Design basis steam generator sodium/water reaction

E-7 One loop natural circulation heat rejection from initial two loop operation

E-8 Rupture disk failure in SGS sodium/water protection system

E-9a Water/steam side isolation and dump of an evaporator/ superheater module with failure of a module outlet isolation valve to close
E-9b Water/steam side isolation and dump of an evaporator/ superheater module with failure of an evaporator inlet isolation valve to close
E-9c Water/steam side isolation and dump of an evaporator/ superheater module with failure of a superheater inlet isolation valve to close

E-l0 Water side isolation of an evaporator module with failure of the water pump valve to open

E-l1 Steam side isolation of a superheater with failure of one relief valve to open

E-12 (Deleted)

E-13a (Deleted)
E-13b (Deleted)

E-14 Inadvertent dump of intermediate loop sodium

E-15 DHRS activation 24 hours after scram

E-16 Three loop natural circulation

E-17 Two loop natural circulation heat rejection from initial three loop operation

E-18 Two loop natural circulation

E-19 Loss of flow in two sodium loops

D. **Faulted Events**

F-1 (Deleted)

F-2 DHRS Activation without SGS cooldown

F-3 a Feedwater line rupture between steam drum and inlet isolation valve
F-3b Feedwater line rupture in main incoming header

F-4a Saturated steam line rupture
F-4b Main steam line rupture

F-4c Rupture between superheater module outlet and superheater outlet isolation valve
F-4d Rupture between superheater outlet isolation valve and main steam line

F-5a Recirculation line break between drum and recirculation pump inlet
F-5b Recirculation line break between evaporator outlet and drum inlet

F-6 Intermediate Loop Sodium-Air Leak