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소듐냉각고속로 정책개발, 기술기획 및 인허가 방안 마련

Policy development and technical planning evaluation for Gen IV SFR

소듐냉각고속로 해외협력 및 인허가 방안 구축

Establishment of the international collaboration and licensing preparation planning for the specific design of a prototype SFR

한국원자력연구원

교 육 과 학 기 술 부

제 출 문

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본 보고서를 “소듐냉각고속로 정책개발, 기술기획 및 인허가 방안 마련”과제 (세부과제 “소듐냉각고속로 해외협력 및 인허가 방안 구축”) 의 보고서로 제출합니다.

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보고서 요약서

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○ 실험로 등을 통하여 설계 및 운전 경험을 보유하고 있으며 기술 개발 경험이 풍부하며 특히 금속연료 기술 및 관련 데이터를 확보하고 있는 미국 ANL과의 개념설계 검토 등을 포함한 중장기적인 협력을 통해 사업단 목표 효과적으로 달성
○ 규제기관의 원형로 인허가 업무 능력 및 효율 향상 도모하기 위하여, 국내 인허가 관련 법령 검토 및 원형로 특정설계 인허가 방안(안) 도출. 미국, 일본, 프랑스, 중국, 인도, 러시아 등 개발 국가의 인허가 검토
○ 제4세대 소듐냉각고속로 기술개발사업을 위한 예비타당성 분석 및 옵션 스터디를 통하여 향후 기술개발 가능성에 대한 신기술 적용 타당성에 대한 연구 수행 및 장기에너지 수급계획 및 원자력발전에 대한 정책 방안 마련</td></tr>
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요 약 문

Ⅰ. 제 목

소듐냉각고속로 해외협력 및 인허가 방안 구축

Ⅱ. 연구개발의 목적 및 필요성

2008년 12월 제255차 원자력위원회에서는 파이로 건식처리와 연계한 소듐냉각고속로 개발을 포함하는 “미래원자력시스템 개발 장기추진 계획”을 심의 의결하였으며, 2011년 3월 제258차 원자력위원회에서는 명확하게 제시된 목표에 따라 종합적인 기술개발 계획 수립과 경영 관리가 가능하도록 소듐냉각고속로 개발을 사업단 체제로 운영할 것을 심의 의결하였다. 이에 따라, 소듐냉각고속로 원형로를 ‘28년에 건설하기 위해 ’20년까지 특정설계인가를 획득하는 것으로 추진하고 있다.

따라서, 위와 같은 목표를 효과적으로 달성하기 위해서 미국, 일본, 프랑스, 중국, 인도, 러시아 등 소듐냉각고속로 개발기관과의 해외렵력 방안 구축하고, 국내 인허가 관련 법령을 검토하여 원형로 특정설계 인허가 방안(안)을 사전에 도출하며, 이의 기반이 될 제4세대 소듐냉각고속로 기술개발사업을 위한 예비타당성 분석 및 옵션 스터디를 수행하게 되었다.

Ⅲ. 연구개발의 내용 및 범위

○ 해외협력 및 원형로 특정설계 인허가 준비
  - 소듐냉각고속로 원형로 설계를 위한 해외협력 방안 구축
  - 미국 ANL과의 기술협력을 통한 개념설계 Review
  - 소듐냉각고속로 원형로 특정설계 인허가 기반 마련 방안 구축

○ 예비타당성분석 및 옵션 스터디
  - 제4세대 소듐냉각고속로 기술개발사업을 위한 예비타당성 분석 및 옵션 스터디

Ⅳ. 연구개발결과

해외협력 강화와 입증기술에 대한 도입을 통해 개발기간을 단축하고 2012년 개념설계 및 2013년 예비특정설계에 대한 국내·외 전문가 검토를 통하여 설계개념을 조기 확정하고 기술검증계획 및 핵연료 개발계획을 구체화하도록 하였다.

ANL(미국)과의 기술협력을 위해서는 WFO Agreement를 체결하였으며, TerraPower(미국)

와의 기술협력, IGCAR(인도)과의 기술협력을 위해 MOU를 체결하였다.

금속연료 장전 SFR 설계경험이 풍부한 ANL과의 양자 계약을 통해 개념설계 및 요건에 대한 자문을 수행하였다. 인도의 SFR 최고 전문가(S. C. Chetal, 인도 IGCAR 전 원장)를 상시 초청하여 인도의 PFBR, FBTR 설계경험을 공유하고 인도 SFR 인허가 시 도출된 주요 인허가 현안을 공유하여 국내 인허가 추진 시 반영할 예정이다.

정례화된 국제전문가자문회의를 통하여 예비특정설계, 기술검증계획 및 절차, 핵연료개발 계획을 보다 구체화하고 참여국가에서 보유한 설계기술, 실증시험시설, 설계인력 등을 공동 활용할 수 있는 방안을 도출하여 단축가능한 개발일정을 수립할 예정이다.

원형로 특정설계 인허가 방안 구축을 위해서도 원자력안전기술원 및 원자력안전위원회와의 업무협의를 진행하였으며, 위탁연구를 통하여 수행한 소듐냉각고속로 원형로 개발을 위한 예비타당성 분석 및 에너지 MIX 옵션 스터디 결과를 추후 지속적으로 반영해 나갈 예정이다.

Ⅴ. 연구개발결과의 활용계획

해외협력 방안 구축을 통하여 채득한 방안에 따라서 향후 협력을 진행함으로서 국내 기술개발의 효율성을 제고하고 국내 소듐냉각고속로 건설 목표를 달성할 수 있도록 활용할 것이다. 또한, 개념설계 검토 및 기술 개발 경험을 이용하여 국내 원형로 설계의 신뢰도를 향상시킬 수 있도록 활용할 예정이다.

인허가 방안 구축을 통하여 국내 원형로 특정설계 인허가 법규를 정비하고 규제기관으로 하여금 인허가 업무를 효율적으로 진행할 수 있는 기반을 조성하고, 소듐냉각고속로 개발 사업의 사전 예비타당성을 평가함으로서 향후 정부의 예비타당성 평가에 대비할 것이다.

초임계 CO_2 브레이튼 사이클 SFR 적용 타당성 연구를 통하여 근본적으로 소듐-물 반응으로부터 자유로운 계통 구성 가능성을 검토하여 안전성과 경제성이 향상된 SFR 계통 구성 타당성을 검토할 것이다.

SUMMARY

Ⅰ. Project Title

Establishment of the international collaboration and licensing preparation planning for the specific design of a prototype SFR

Ⅱ. Objectives and Necessity of the Study

For the successful development of Sodium cooled fast reactor(SFR) in Korea, Sodium cooled Fast Reactor development Agency(SFRA) was created in May 2012, with the objective of obtaining the design and construction license for the final construction of PGSFR(Prototype Gen IV SFR) by 2028.

International collaboration with the advanced countries in SFR development, such as USA, Japan, France, India, China, Russia and France would be essential for the effective research and development of SFR in Korea. And the establishment of the licensing procedure for the prototype reactor would be necessary before the submission of documents to the licensing authorities.

Ⅲ. Contents and Scope of the Study

○ International collaboration and the preparation for the licensing of the prototype reactor
  - International collaboration for design of SFR prototype reactor
  - Technical collaboration with ANL for the review of conceptual design
  - Preparation for the licensing procedure for the SFR prototype reactor

○ Pre-applicaion of the feasibility study and energy mix option study
  - Pre-application of the feasibility study for the Gen IV SFR technology development and energy mix option study

Ⅳ. Results of the Research and Development

The conceptual design of prototype of Gen IV SFR (PGSFR) will be early determined through the review of the international experts. After this, the technology demonstration plan and validation of fuel design will be determined in more detail. The project will be accomplished efficiently by introducing the proven technology already validated from the international collaboration.

The conceptual design and its requirements of PGSFR will be reviewed by ANL, who has a lot of design experiences in the metal fueled SFR development. The collaboration with ANL has been done through Work For Others (WFO) contract, and the MOU was signed between SFRA and TerraPower(USA), and SFRA and IGCAR.

The licensing issues raised during PFBR and FBTR licensing in India will be discussed and reflected into the PGSFR design by inviting the high level expert from India, for example Dr. Chetal in IGCAR.

The specific design, technology validation plan and fuel development plan will be established in more detail through the annual International Technical Review Meeting (ITRM) and experimental facilities available from the international institute and companies, which will be the basis for shortening the project period and to reduce the development cost.

Ⅴ. Future Application of the Study

The established project plan will be more efficiently accomplished by international collaborations with the advanced countries in SFR development areas. And the reliability of our design will be much improved by using the international technical review and the experiences in the research and development in SFR.

And the establishment of the licensing procedure for the prototype reactor will be used as a basis for the submission of documents to the licensing authorities. This project will be prepared to the feasibility review by the pre-application of the feasibility study.

The commercialization of sodium cooled fast reactor will be pursued through safety enhancement and improved economy, from the feasibility study of supercritical CO2 Brayton cycle linked to SFR, fundamentally free from Na-water reaction.

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제 1 장 연구개발과제의 개요

제 1 절 연구개발의 필요성

2008년 12월 제255차 원자력위원회에서는 파이로 건식처리와 연계한 소듐냉각고속로 개발을 포함하는 “미래원자력시스템 개발 장기추진 계획”을 심의 의결하였으며, 2011년 3월 제258차 원자력위원회에서는 명확하게 제시된 목표에 따라 종합적인 기술개발 계획 수립과 경영 관리가 가능하도록 소듐냉각고속로 개발을 사업단 체제로 운영할 것을 심의 의결하였다. 이에 따라, 소듐냉각고속로 원형로를 ‘28년에 건설하기 위해 ’20년까지 특정설계인가를 획득하는 것으로 추진하고 있다.

따라서, 위와 같은 목표를 효과적으로 달성하기 위해서 미국, 일본, 프랑스, 중국, 인도, 러시아 등 소듐냉각고속로 개발기관과의 해외렵력 방안 구축하고, 국내 인허가 관련 법령을 검토하여 원형로 특정설계 인허가 방안(안)을 사전에 도출하며, 이의 기반이 될 제4세대 소듐냉각고속로 기술개발사업을 위한 예비타당성 분석 및 옵션 스터디를 수행하게 되었다.

본 연구의 경제 산업적 중요성과 기술개발 필요성은 다음과 같다.

○ 경제·산업적 중요성

- 소듐냉각고속로(SFR)는 우라늄 자원의 3%밖에 사용할 수 없는 경수로와는 달리 우라늄자원 이용률을 획기적으로 향상시켜 지속가능한 에너지 자원을 확보할 수 있으며, 또한, 국내 소비 에너지 자원의 97%를 수입에 의존하고 있는 우리나라의 경우 소듐냉각고속로 도입으로 인하여 안정적인 에너지 자원을 확보하는 매우 큰 경제적 효과가 있음

- 소듐냉각고속로는 사용후연료를 재순환하여 사용함으로써 경수로의 사용후연료 처분문제를 해결할 수 있는 원자로시스템으로서 조기에 실현 가능한 유일한 대안으로 인정받고 있으며, 이를 통해 고준위 폐기물의 영구처분량을 획기적으로 감소시켜 기존 원자로보다 환경친화성이 탁월할 뿐만 아니라 사용후핵연료의 저장 및 처분시설의 용량을 획기적으로 줄여주어 환경친화성과 대중 수용성을 높일 수 있음

- 제255차 원자력위원회에서는(2008년 12월) 파이로 건식처리와 연계한 소듐냉각고속로 개발을 포함하는 “미래원자력시스템 개발 장기추진 계획”을 심의 의결하였으며, 2011년 3월 제258차 원자력위원회에서는 명확하게 제시된 목표에 따라 종합적인 기술개발 계획 수립과 경영 관리가 가능하도록 소듐냉각고속로 개발을 사업단 체제로 운영할 것을 심의 의결하였음. 이에 따라, 소듐냉각고속로 원형로를 ‘28년에 건설하기 위해 ’20년까지 특정설계인가를 획득하는 것으로 추진하고 있음

○ 기술개발 필요성

- 위와 같은 사업 목표를 효과적으로 달성하기 위해서는 원형로 개발을 통한 국내 고속로

설계기술 축적 및 실증이 필요하며, 기술개발 시기를 앞당기고 사업단의 목표를 효과적으로 달성하기 위해서는 소듐냉각고속로 개발 해외 선진국과의 긴밀한 협력이 필요하며 이를 위한 방안을 도출하는 것이 요구됨

- Fermi, FFTF 등의 소듐냉각고속로 실험로와 IFR 등 기술개발 경험이 풍부하며 특히 금속연료 기술 및 관련 데이터를 확보하고 있는 미국 ANL과의 협력은 국내 금속연료 기술 개발에 반드시 필요하여, 개념설계 검토 등을 포함한 중장기적인 협력을 추진
- 또한, 국내 법규상 미비한 소듐냉각고속로 특정설계의 인허가 방안을 사전 검토하여 규제기관이 법제화하는데 기초자료를 생성하고, 궁극적으로는 인허가 업무 효율 향상을 도모하여 효과적인 인허가 획득을 기하고자 함

본 연구의 기술개발 범위는 다음과 같이 크게 네가지로 구분할 수 있다.

- 소듐냉각고속로 개발 해외협력 방안 구축
- KAERI-ANL 기술협력 프로그램
- 소듐냉각고속로 원형로 특정설계 인허가 방안 구축
- 제4세대 소듐냉각고속로 기술개발사업을 위한 예비타당성 분석 및 옵션 스터디

제 2 절 연구개발 목표 및 내용

1. 최종목표

제4세대 소듐냉각고속로 해외협력 방안 및 원형로 특정설계 인허가 기반 마련 방안 구축

2. 연구 목표 및 내용

구분	년도	연구개발목표	연구개발내용
1차년도	2012	해외협력 및 원형로 특정설계 인허가 준비	- 소듐냉각고속로 원형로 설계를 위한 해외협력 방안 구축 - 미국 ANL과의 기술협력을 통한 개념설계 Review - 소듐냉각고속로 원형로 특정설계 인허가 기반 마련 방안 구축
		예비타당성분석 및 옵션 스터디	- 제4세대 소듐냉각고속로 기술개발사업을 위한 예비타당성 분석 및 옵션 스터디

제 2 장 국내·외 기술개발 현황

제 1 절 국내 기술개발 현황

우리나라는 한국원자력연구원을 중심으로 기초기술 연구를 시작한 이래로 1997년부터 국가원자력연구개발 중장기계획사업을 통해 본격적으로 소듐냉각고속로 연구개발에 착수하여, 2001년에 소형 소듐냉각고속로인 KALIMER-150(150MWe)의 개념설계를 완성하였다.

2002년부터 2006년까지는 우리 기술력을 바탕으로 한 독창적 개념의 중형 소듐냉각고속로인 KALIMER-600(600MWe)의 개념설계를 완성한 바 있다. KALIMER-600 원자로 개념은 우리 기술력을 바탕으로 한 독창적 개념의 원자로로서 일본의 JSFR과 함께 상용화 가능성 입증을 위한 제4세대 소듐냉각고속로의 참조 노형으로 선정되기도 하였다.

KALIMER-600 개념설계 경험을 바탕으로 2007년부터 제4세대 소듐냉각고속로 고유개념 설정을 위한 연구를 수행한 바 있으며, 이 과정에서 Gen IV 4대 기술목표를 달성하기 위해 제시된 복수 후보개념에 대하여 설계사양에 대한 기술적 타당성 평가를 수행하였다.

제4세대 원자력시스템으로서 기술목표를 달성하기 위해 제시된 복수 후보개념에 대하여, 용량, 노심형태, 피복재 합금후보, barrier 후보, 루프수, 증기발생기 전열관 형식 등의 설계사양에 대한 기술적 타당성 평가를 통해 단일 고유개념으로서의 최적후보개념을 도출하고 설계사양을 설정하였다.

최적후보개념에 대한 종합적 평가를 통하여 자체순환로의 경우 용량 1,200MWe, 농축도 분리노심, 2-루프, 이중벽 증기발생기, 원자로내부 가동중 검사 장수명 센서 시스템 등의 고유개념을 설정함. 연소로의 경우에는 용량 600MWe인 KALIMER-600의 설계개념을 기초로 하여 단일농축도 노심장전모형을 개발하여 고유개념으로 설정한 바 있다.

또한 설정한 고유개념의 성능달성을 위하여 최상위 설계요건을 도출하고 설계기준 수립하여 개념설계 방향을 제시하였다.

2008년 12월 22일 제255차 원자력위원회에서 “친환경 고속로 순환핵연료주기시스템 개발 장기 추진계획”을 의결하고[그림 1], 그에 따라 2028년 제4세대 소듐냉각고속로 실증로 건설을 위해 2011년 개념설계 완료, 2020년 표준설계승인을 목표로 설정하였다.

이후 2011년 11월 제1차 원자력진흥위원회에서 가용예산 범위내에서 원형로(100MWe급) 건설을 추진하고 실증로 건설 없이 상용로를 개발하도록 “미래원자력시스템 개발 장기추진 계획” 수정·의결하였으며[그림 2], 위에서 제시된 목표에 따라 종합적인 기술개발 계획 수립과 경영·관리가 가능하도록 사업단 구성을 의결하고, 2012년 5월 16일 명확한 사업목표 달성을 위해 성과중심의 개방형 소듐냉각고속로개발사업단(이하 사업단)을 출범하였다.

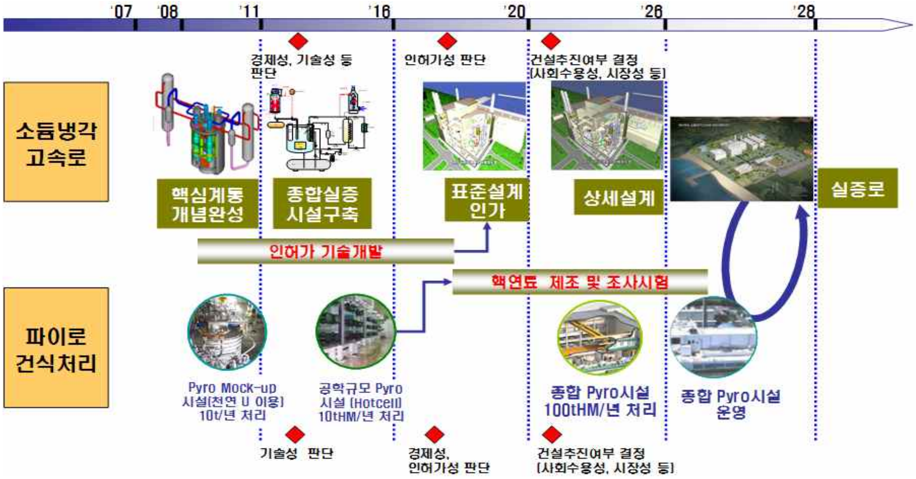


그림 1. 친환경 고속로 순환핵연료주기 시스템 개발 장기 추진계획

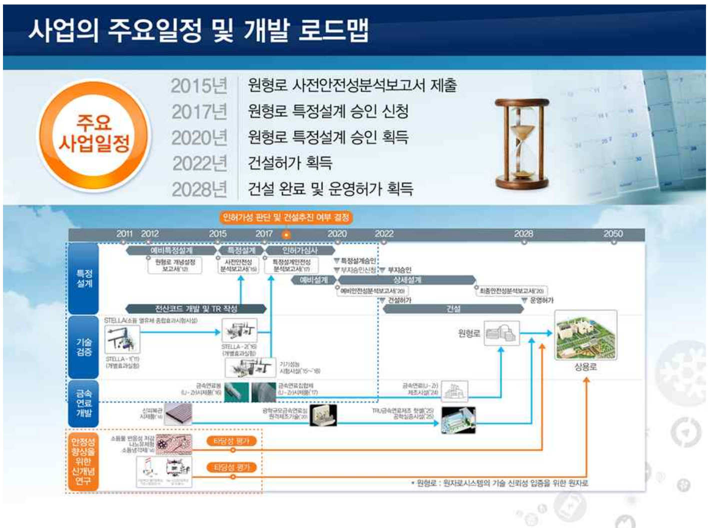


그림 2. 친환경 고속로 순환핵연료주기시스템 개발 장기 추진 수정계획

2012년에는 사업단을 중심으로 전문가의 기술검토를 거쳐 2028년 건설 목표로 개발중인 원형로의 용량을 150MWe로 결정하였으며, 개념설계를 위한 최상위설계요건을 설정하고, 이를 만족하는 개념설계를 완료하였다.

그러나, SFR의 개념설계는 경험과 기술을 어느 정도 확보하였으나 개념설계 기술과는 별도로 검증실험 등의 하드웨어 기술은 소규모에 그치고 있으며, 주요기기 제작기술 및 BOP 관련 기술과 인허가 기술 등은 아직 시작하지 못하여 매우 미흡한 상태에 있다.

기본설계와 상세설계 등과 특히 BOP 설계 분야에서는 해외의 경험을 활용하여 기술 개발 비

용과 기간을 단축할 수 있을 것으로 기대하고 있으며, 국내 개발 SFR 원형로에 대한 인허가 법령이 미비하여 해외 사례 조사를 통하여 구축할 필요가 있다.

또한, 제4세대 소듐냉각고속로 기술개발사업을 위한 예비타당성 분석 및 옵션 스터디를 통하여 향후 기술개발 가능성에 대한 신기술 적용 타당성에 대한 연구를 수행하며, 장기에너지 수급계획 및 원자력발전에 대한 정책 방안을 마련할 수 있을 것으로 기대하고 있다.

원자력 선진국인 일본, 러시아, 프랑스 등과 에너지 다소비국인 중국, 인도 등은 소듐냉각고속로의 실용화 기술개발을 2030년경까지는 완료할 계획으로 우라늄 자원의 한계성을 극복하고, 안전성과 환경보호 측면이 대폭적으로 강조된 개념을 개발하고자 노력하고 있는 등 향후 소듐냉각고속로는 경수로 이후의 주종노형으로서 전력공급을 대체해 나갈 것이다.

에너지자원이 부족한 우리나라로서는 전력공급은 물론 사용후핵연료의 안전한 처리를 위해서도 소듐냉각고속로의 도입은 필연적이며, 자체개발 및 국제공동연구를 통하여 핵심기술 확보가 이루어질 것이다. 우리나라에서 소듐냉각고속로 설계기술 개발을 위해서는 경·중수로 설계기술의 70~80%를 전용할 수 있는 것으로 분석되고 있으며, 그간 국내에 기술축적이 되어온 경수로 NSSS 계통설계, 핵연료설계를 통하여 축적된 설계기술 및 인력을 활용할 수 있을 것이다.

소듐냉각고속로의 고유안전특성, 핵확산 저항성을 보유한 핵연료주기, 고준위 방사성 폐기물 발생량의 저감 등이 기술적으로 입증될 것이며, 앞으로 예상되는 20여 년간의 경제성 입증기간 동안에 신기술 및 혁신설계개념의 도입으로 경제성 향상을 위한 노력이 집중될 것이다.

소듐냉각고속로 기술개발은 장기적이고 체계적인 계획 하에 추진하는 것이 필요하며, 핵확산 저항성, 안전성 및 경제성을 높일 수 있는 전략 핵심기술의 개발이 필요함. 현재 국내에서의 노력은 선진국에 비해 비록 출발점은 상당히 뒤져 있는 실정이나, 단계적인 전략을 세워 우선순위에 따라 개발 노력을 집중함으로써 국내 소듐냉각고속로 기술은 획기적으로 발전될 수 있을 것으로 전망된다. 소듐냉각고속로 기술은 핵확산의 우려에 따라 핵심기술의 이전·도입이 극히 제한되기 때문에 국제협력을 통하여 기술 개발의 투명한 신뢰관계를 구축하여 각 세부분야의 부족기술은 해외에서 도입하되 핵심기술 부분은 국내에서 기술 개발을 추진하는 것이 바람직할 것으로 보인다.

2012년부터 시작되는 원형로 개념설계 단계부터 산업체의 부분적인 참여가 필요하며, 이후 특정설계부터는 주증기공급계통(NSSS) 분야는 KAERI가 동력변환계통(BOP) 분야와 기기 제작성 검토는 산업체가 맡아 상호 협력하여 원형로를 개발하는 것이 필요하다.

제 2 절 국외 기술개발 현황

1. 국외 주요 연구동향

소듐냉각고속로는 현재까지 전 세계적으로 약 400 원자로·년 이상의 운전실적을 보유하고 있으며 제4세대 원자력시스템 중에서 가장 빠른 상용화가 기대되는 노형으로 주목받고 있다.[그림 3]

그림 3. 소듐냉각고속로 해외 개발 현황

국내·외 전문가들은 쌓여만 가는 사용후핵연료 처리를 위해서는 소듐냉각고속로의 개발을 앞당기는 것이 기술적으로 가장 타당하다고 평가하고 있다.[그림 4]

후쿠시마 원전사고 이후 세계원자력 시장의 관심은 경제성에서 안전성과 사용후핵연료 문제로 변화하고 있으며, 기존의 원자력 선진국들은 '20년대 기술실증을 목표로 기술개발을 진행하고 있다. 러시아, 인도, 중국등 신흥경제국(BRICs)들은 '20년대 상용로 도입을 통해 기득권 세력이 있는 경수로 시장보다, 기술선점이 가능한 미래원자력시장을 목표로 기술개발에 박차를 가하고 있다.[그림 5]

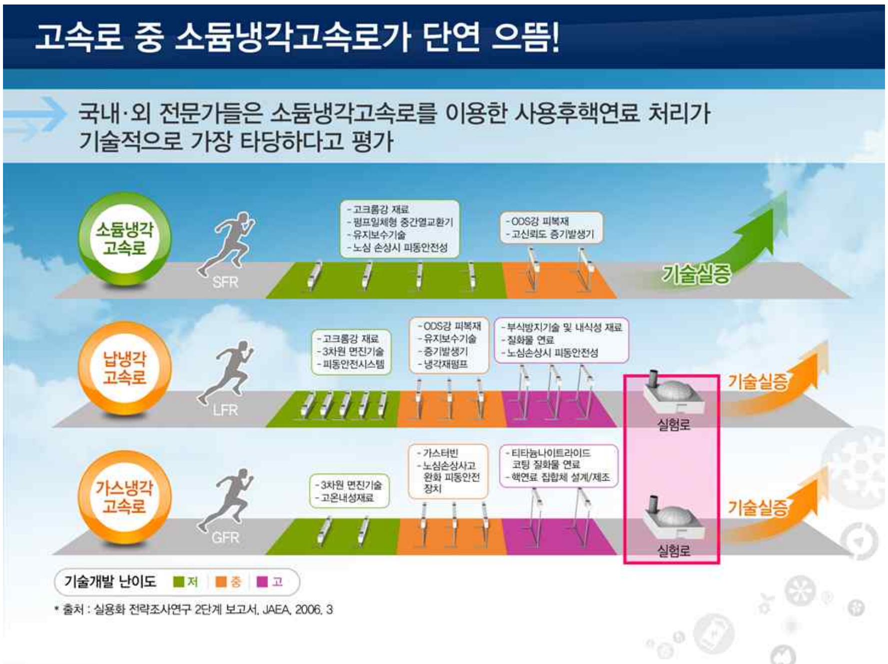


그림 4. 제4세대 원자로 경쟁력 비교

그림 5. 원자력 발전시장의 패러다임 변화

주요 원자력 이용국들의 소듐냉각고속로 개발 현황은 다음과 같다.

○ 일본: 2050년대 상용화

- Joyo(50~140MWth, 실험로): ‘77년 초임계, ’08 시험집합체 손상으로 일시 중지
- Monju(280MWe, 원형로): ‘94년 초임계 달성, ’10년 핵연료 교환장치 수리 후 재가동 대기중
  . 크고 작은 고장과 사고로 인하여 건설 목적을 달성하지 못함
  . ’95년 12월 소듐 누설로 인한 화재 발생 및 처리
  . ’10년 8월 원자로내 핵연료 이송장치 낙하사고 발생 및 후속조치
  . 향후 지속적인 SFR 개발을 위해 재가동 대기중
- 실증로(500~750MWe): ’25년경 건설완료 예정
- 상용로(1,500MWe): ’50년경 도입 계획

○ 프랑스: 2040년대 상용로 도입 계획

- Rapsodie(40MWth): '67. 1 ~ '83. 10
- Phenix(250MWe): '73. 8 ~ 2009. 3
- Super-Phenix(1,200MWe): '85. 9 ~ 98. 2
  . '72년도 우라늄가격 상승에 대비한 프랑스, 독일, 이탈리아 3국의 건설 계획
  . '85년 가동시 경기침체로 인한 원자력 수요 감소로 우라늄가격 변동 없었음
  . '86년 체르노빌사고 영향, 사회당과 녹색당 연정 후 녹색당의 선거공약 등으로 '96년 가동중지 후 '98년 폐쇄
  . 약 11년 가동 중 운전 4.5년, 보수 2년, 재가동 허가 4.5년 소요됨
  . 출력이 Phenix(250MWe)의 약 5배인 1,240MWe으로 급격한 용량 증대에 따른 경험 부족과 프랑스, 독일, 이탈리아가 각자 설계하고 제조함으로서 기술 기준 차이 및 연계성 부족 등에 의한 문제점 발생
  . 극복 불가능한 근원적인 문제는 아니었음
  . 충분한 준비 없이 추진된 대규모 국제공동 건설 사업의 실패사례로 기록됨
- ASTRID(600MWe, 원형로): ‘22년 건설완료 예정

○ 러시아: 2020년대 상용화

- MBIR(40MWe, 실험로): ‘19년 운전 예정
- BN-800(880MWe, 실증로): ‘14년 건설완료 예정
- BN-1200(1,200MWe, 상용로): ‘20 건설완료 예정 (’12년 건설허가 획득)

○ 중국: 2030년대 상용화

- CEFR(20MWe, 실험로): ‘10년 건설완료
- CFR-600/1000(600MWe/1,000MWe, 실증로): ‘25년 건설완료 예정

○ 인도: 2020년대 상용화

- FBTR(13.5MWe): ‘85년 건설완료
  . ’20년 수명 (수명 40년으로) 연장 (‘11. 2)
  . ’13년 금속연료로 대체
  . 내부검사를 위한 잠망경 개발
- PFBR(500MWe, 원형로): ’13년 건설완료 예정
- CFBR(500MWe, 상용로 6기): ‘23년 건설완료 예정
- FFBR(1,000MWe, 상용로): '25년 이후 건설 계획

2. 국내외 연구현황

국외 소듐냉각고속로 기술수준은 현재 건설·운전·폐쇄 15기(실험로: 11, 원형로: 3, 실증로: 1), 가동중 6기(실험로: 4, 원형로: 2), 건설중 2기(원형로: 1, 실증로: 1)이며, 국내외 소듐냉각고속로 연구개발 현황을 표로 요약하였다. [표 1]

표 1. 국내외 소듐냉각고속로 연구개발 현황

연구수행 기관	연구개발의 내용	연구개발성과의 활용현황
KAERI, 한국	-제4세대 소듐냉각고속로 설계 및 기술검증 -제4세대 소듐냉각 고속로 개념설계 -종합 열유체 효과 검증실험 -금속연료 개발 -신개념 연구 개발	-제4세대 소듐냉각고속로 원형로 특정설계 및 인허가 획득에 활용
ANL, INL, PNL 등 미국	-1951년 실험로 EBR-I(200kWe) 건설, 1963년 실험로 EBR-II(20MWe) 건설 -사용후연료 관리를 위한 TRU 연소로 설계 및 개발(ABTR) -고속로 핵연료 기술개발	-소듐냉각 고속로 계통설계 기술 개발과 건설을 위한 기반 기술 확보 -IFR 설계 개발 및 PRISM 원자로 개발에 활용 -ABTR (Advanced Burner Test Reactor) 노심 예비개념 설계 -미국 ABTR 소듐냉각 고속로에 초임계 CO_2 Brayton Cycle 동력전환계통을 적용 -MA/RE 함유 조사시험 금속연료 제조하여 ATR 연구로를 이용하여 연료특성 평가를 위한 조사시험 수행 -금속연료봉 설계기술 확보 및 핵확산저항성에 대한 기술타당성 평가 -HT9 피복관/덕트 조사성능 자료 확보
CEA, 프랑스	-임계시설(MASURCA) 및 원형로(Phenix) 운영 -300 MWe급 원형로 Phenix를 설계 변경하여 수명 종료 시험을 통해 안전성 평가 전산코드 검증 -ASTRID 혁신 설계 개념 연구	-MASURCA를 이용한 노심 개념 검증 및 노심 핵설계 전산코드 검증 -원형로(Phenix) 운영을 통한 소듐냉각 고속로 노심 운전 능력 확보 -소듐냉각 고속로 혁신 설계 개념 개발 및 차기 원형로(ASTRID) 설계 반영 -EFR 설계 개발, ASTRID 개념 설정 -핵연료 취급장치 개선, G91강의 배관 및 기기 재료 채택여부, 혁신적 가동중검사 센서 개발, Advanced SG 개발을 추진하여 2012년 개념설계에 반영 계획 -소듐냉각 고속로에 초임계 CO_2 Brayton Cycle 동력전환계통 개념의 적용성 연구를 수행중
JAEA & CRIEPI, 일본	-MONJU(원형로) 재가동 -고속로주기 실용화 전략 조사연구 (FaCT 프로젝트) -고속로 핵연료 기술개발 -JSFR 적용을 위한 혁신적 기술개발	- MONJU 핵연료이송장치 수리 후 재가동 준비중 - 2025년경 실증로 건설에 반영 - 경제성 안전성 향상을 위한 상용로 노심 설계 개념 개발 -FaCT(FBR 사이클 실용화 연구개발)를

연구수행 기관	연구개발의 내용	연구개발성과의 활용현황
		통하여 전력중앙연구소(CRIEPI)가 주도적으로 금속연료 개발 -고크롬강 채택으로 배관길이 단축, 통합 IHX-Pump 개발, 원자로용기 소형화, 핵연료취급장치 개선, 이중배관 적용기술, 이중벽관 SG 개발, 소듐중 가동중검사 기술 개발, 및 3차원 면진설계기술을 개발하여 소듐냉각고속로의 안전성과 경제성 향상 구현
IPPE, 러시아	- BFS-1, BFS-2 임계실험시설 및 BOR-10, BOR-60, BN-600 운영 -소듐냉각 고속 실증로 설계 및 건설	-설계 개념 검증 및 노심 핵설계 코드 검증 자료 생산 -BN-800 건설중
IGCAR, 인도	-FBTR(15MWe급) 실험로 운전 -PFBR(500MWe급, 원형로) 건설 2013년 완료 예정	-PFBR 원자로 개발에 활용 -후속 상용 소듐냉각 고속로 설계개념 개발에 활용

제 3 장 연구개발수행 내용 및 결과

제1절 연구개발 추진전략·방법 및 추진체계

1. 연구개발의 추진전략·방법

○ 소듐냉각고속로 개발 해외협력 방안 구축

– 해외 기술 협력을 통해 소듐냉각고속로 개발 선진국의 설계 경험을 활용하고, 실험 자료를 입수하여 국내 검증용 자료로 활용함

– 해외 실험시설 또는 고속로를 이용한 핵연료 및 피복재 등 재료 조사 실험을 수행할 수 있는 기반을 구축함

○ KAERI-ANL 기술협력 프로그램

– 소듐냉각고속로에 대한 설계 및 운전 경험이 풍부한 미국 ANL과의 협력을 통하여 국내 기반기술을 확보함

– 개념설계에 대한 ANL측의 검토를 통하여 국내 개념설계의 신뢰성을 제고함

– 금속연료 실험 자료, FFTF 및 EBR-II의 노물리 실험 자료와 열유체 실험 자료들을 확보하여 국내 검증자료로 활용함

– 고속로 핵연료 연소시험 등 전산코드 검증 및 인허가를 위해 필요한 실험자료를 확보하여 국내 검증 및 인허가 자료로 활용함

○ 소듐냉각고속로 원형로 특정설계 인허가 방안 구축

– 소듐냉각고속로 특정설계의 인허가 방안을 사전 검토하여 규제기관의 인허가 업무 효율 향상을 도모하고, 국내 법규상 미비한 소듐냉각고속로 특정설계의 인허가 방안을 사전 검토하여 규제기관이 법제화하는데 기초자료를 생성하고, 궁극적으로는 인허가 업무 효율 향상을 도모하여 효과적인 인허가 획득에 기여함

– 미국, 일본, 프랑스, 중국, 인도, 러시아 등 개발 국가의 인허가 자료 및 국내 인허가 관련 법령을 검토하여 원형로 특정설계 인허가 방안(안) 도출함

– 인허가 방안 구축을 통하여 국내 원형로 특정설계 인허가 법규를 정비하고 규제기관과의 기술교류를 통하여 기술을 공유하여 인허가 업무를 효율적으로 진행할 수 있는 기반을 조성함

– 원형로 특정설계 인허가 업무를 위한 기반을 조성하고 규제기관으로 하여금 소듐냉각고속로 원형로 특정설계 인허가에 대한 대비를 세우도록 함

○ 제4세대 소듐냉각고속로 기술개발사업을 위한 예비타당성 분석 및 옵션 스터디

– 제4세대 소듐냉각고속로 기술개발사업 예비타당성조사 사전 대응 연구

– 초임계 CO_2 브레이튼 사이클 SFR 적용 타당성 연구

– 녹색에너지시대의 장기 에너지 Mix 검토

2. 연구개발의 추진체계

○ 해외협력, KAERI-ANL 기술협력 및 인허가 방안 구축을 통하여 제4세대 소듐냉각고속로 해외협력 방안 및 원형로 특정설계 인허가를 위한 기반 마련 방안 구축 목표를 달성하고자 함

○ 연구개발 추진체계를 그림 6에 나타내었다.

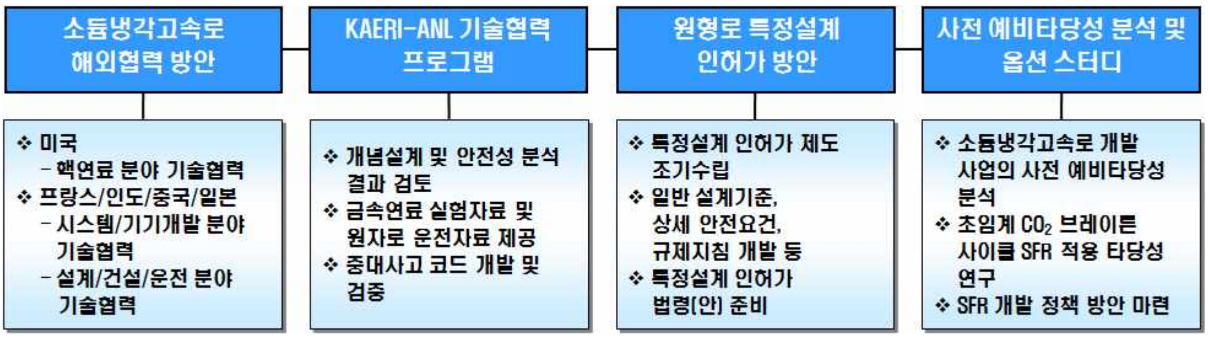


그림 6. 연구개발 추진체계

제2절 연구개발 수행 결과

1. TerraPower(미국) 기술협력 현황 및 계획

○ 목적

– 테라파워와 사업단이 추구하는 기술개념과 방향이 유사하여 협력을 통해 예산/인력 절감 및 일정 단축

○ 경과

– 상호 기술분야 협의('12.6) : 협력 필요성 공감

– 테라파워와 사업단 간 MoU 체결('12.6) [표 2]

– 상호 협력분야(핵연료, 안전해석, 노물리 검증분야 등) 논의('12.7, '12.12)

○ 향후 계획

– 핵연료제조와 피복재 등 재료분야와 스텔라 시험시설을 활용한 공동 기기 검증 분야에 협력을 국한

– 장기적으로 인허가를 위한 시험자료의 공유 등 점진적 확대를 추진

○ 협력 현황

– TerraPower는 Bill Gates재단의 지원을 받아 (초)장주기 소듐냉각고속로를 개발 중에 있으며, 2020년대 상용화를 목표로 하고 있음

· TerraPower는 2007년 Bill Gates (Chairman), Nathan Myhrvold (Vide Chairman), Dr. Gilleland(CEO) 3인에 의해 설립된 원자력 벤처회사로서, Bill Gates는 향후 5년 동안 최대 10억 달러의 자금을 투입할 계획임

· 전세계적으로 모든 나라에서 원자력 안전성, 핵확산 저항성, 에너지 공급 안정성 및 경제적인 전력 공급 향상 달성을 목표로 하고 있으며, 현재는 신개념 원자로 및 원자력 시스템 혁신과 개발의 중심축 역할을 담당하려고 함

– TerraPower는 TWR(travelling wave reactor: 진행파원자로) 개발 프로그램을 진행하고 있음

· 테라파워는 "'TWR'를 10년 가동한다면 생산되는 폐기물의 양은 기존 원자로에서 생산되는 폐기물보다 훨씬 적을 것"이라고 함

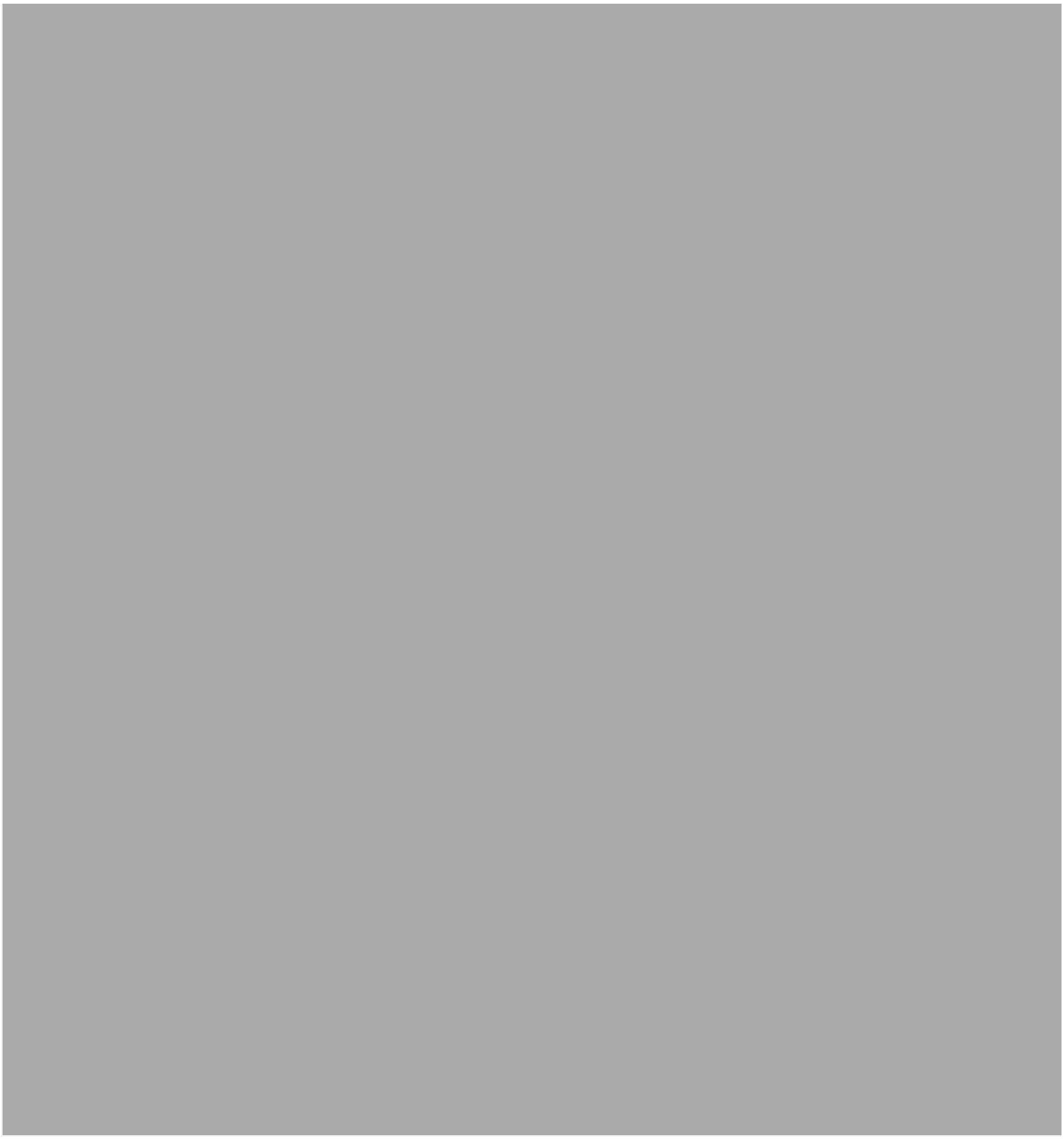
· TWR은 전력 효율성이 높고, 기존의 경수로와 달리 추가적인 연료 보충 없이 100년간 운영될 수 있다는 점에서 주목을 받고 있음.

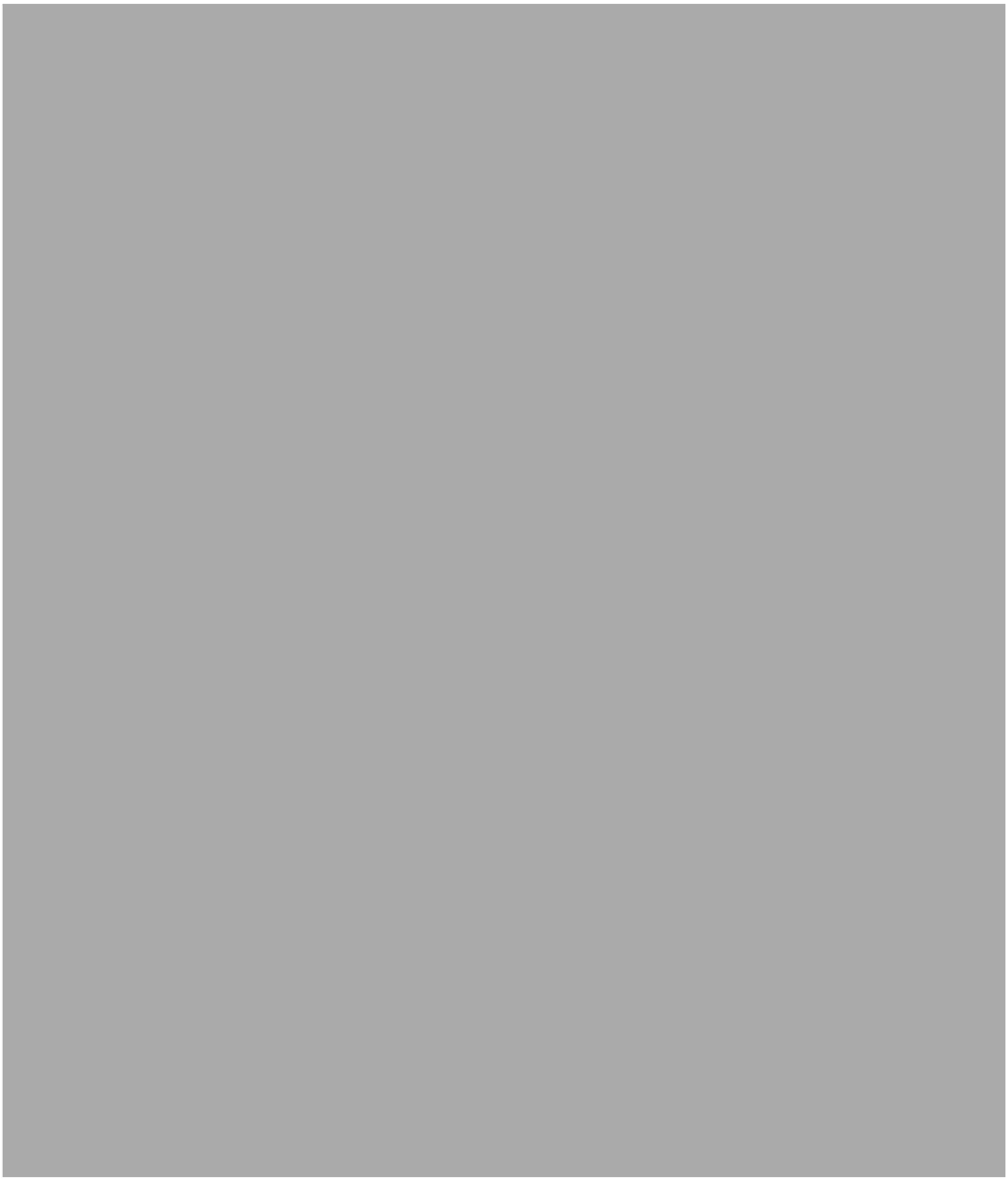
· 또한 농축과정에서 생성된 감손 우라늄을 원료로 사용해 우라늄 농축 시설이 필요

하지 않다는 장점이 있음

- 2012년 6월 5일 테라파워측(사장, 재무부사장, 기술부사장, 기술부장 등 4명)이 사업단을 방문하여 상호 기술분야에 대한 협의를 가짐
  - · 추구하는 기술 개념과 방향의 유사성에 근거하여 긴밀한 협력 필요성을 공감함
- 테라파워측의 MOU 제안을 수용하여 상호 서명하였음[표 2]
  - · 2012년 7월 9일~10일 사업단 2인과 연구부서 2인이 기술토의를 위해 테라파워를 방문하여 핵연료, 안전해석, 노물리 검증 분야 등 상호 협력 분야에 대해 논의하였음
  - · 2012년 12월 13일~14일 사업단 테라파워측 5인이 KAERI를 방문하여 기술협력 회의를 개최함
- 향후계획
  - · 1차적으로 핵연료제조와 피복재 등 재료분야와 스텔라 시험시설을 활용한 공동 기기 검증 분야에 협력을 국한하고, 장기적으로 인허가를 위한 시험자료의 공유 등 점진적 확대를 추진 예정임

표 2 KAERI-TerraPower MOU





2. ANL(미국) 기술협력 현황 및 계획

○ 목적

- 금속연료 장전 SFR 기술 및 실험자료 활용, 원형로 특정설계 수행의 효율성 향상과 결과물의 신뢰도 확보

○ 경과

- 상호협력에 대하여 '11년부터 지속적으로 논의

  · '11.6 한국의 SFR 개발을 위한 ANL 협력에 총괄적으로 합의

  · '12.5 '12년도 협력업무(SFR 원형로 개념설계 리뷰, 특정설계 기술개발과 인허가 획득을 위한 코드 현황 및 검증 실험자료 현황 분석 등) 논의[표 3] 및 '12년 예산 약 20억원(170만불) 확정 [표 4]

- 상호협력에 대한 미국정부 승인('12.11)

  · '12년 예산중 1차 85만불 지급('12.11.13)

- 상호협력에 대한 미국정부(NNSA)의 한국정부 보증 요구('12.12)

  · 한국정부 보증 완료('12.12)

  · '12년 예산중 2차 85만불 지급('13.1.11)

- '13년 ANL 수행업무 도출('13.1)

  · '13년 예산은 약 25억원(208만불) [표 5]

○ 향후 계획

- KAERI와 ANL의 구체적 업무분장, 사용 전산코드 및 방법론 결정, 상세일정 작성은 금년내에 구체적으로 논의

- KAERI 연구진이 ANL을 방문하여 공동 작업을 진행

- 정기적으로 양측을 오가며 진행 상황 점검

○ 협력 현황

- ANL과의 협력으로 금속연료 장전 SFR 기술 및 실험자료 활용, 원형로 특정설계 수행의 효율성 향상과 결과물의 신뢰도 확보를 통해, 2020년까지 예정된 원형로 특정설계 승인의 성공적 획득에 도움이 될 것임

  · ANL은 세계최초로 SFR 실험로 EBR-I을 '51년에 가동한 바 있으며 이후 EBR-II, FFTF 등의 실험로를 건설, 운영한 바 있음

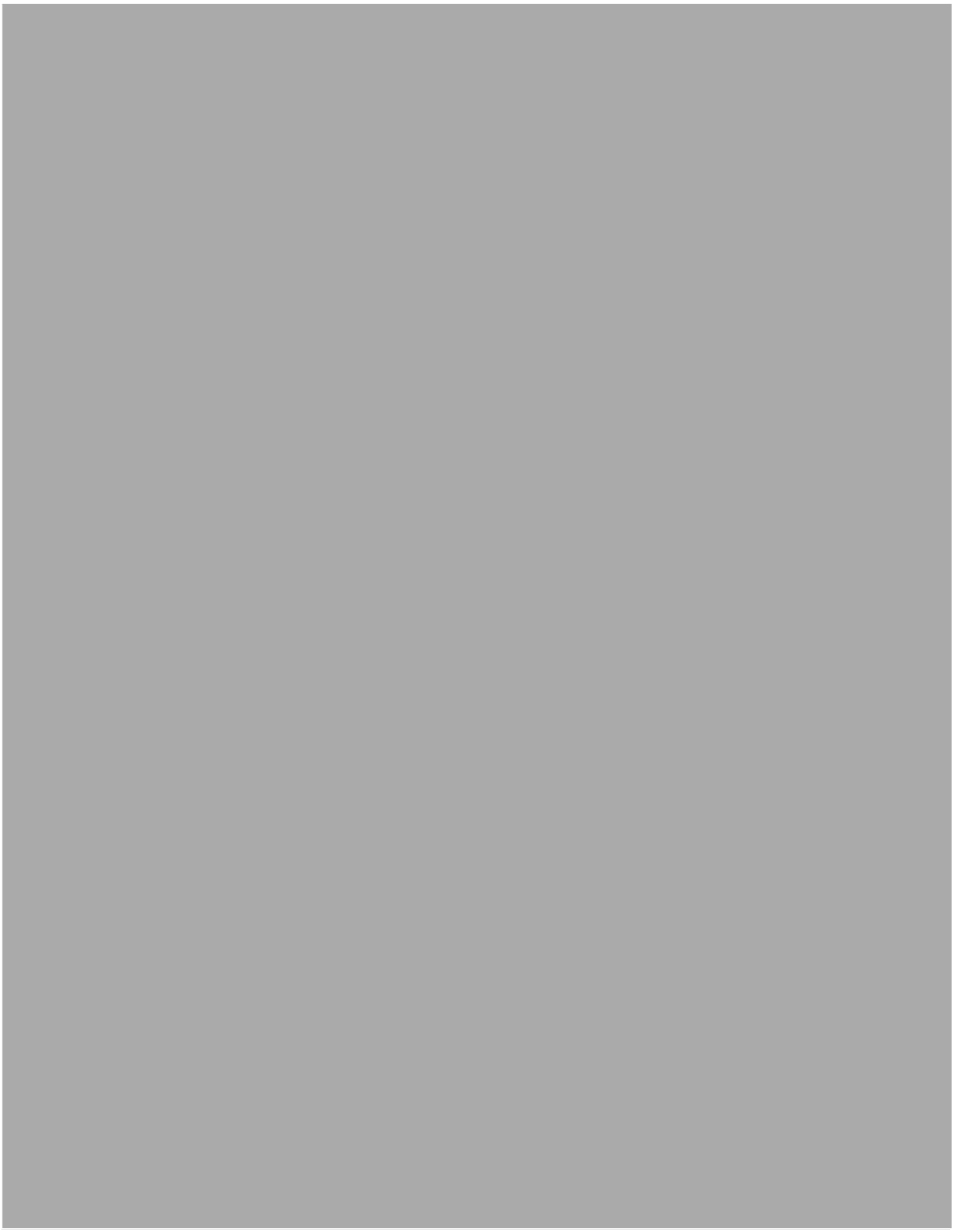
  · 뿐만 아니라 ZPPR 노물리 실험, 고속로 핵연료 연소시험 등 전산코드 검증 및 인

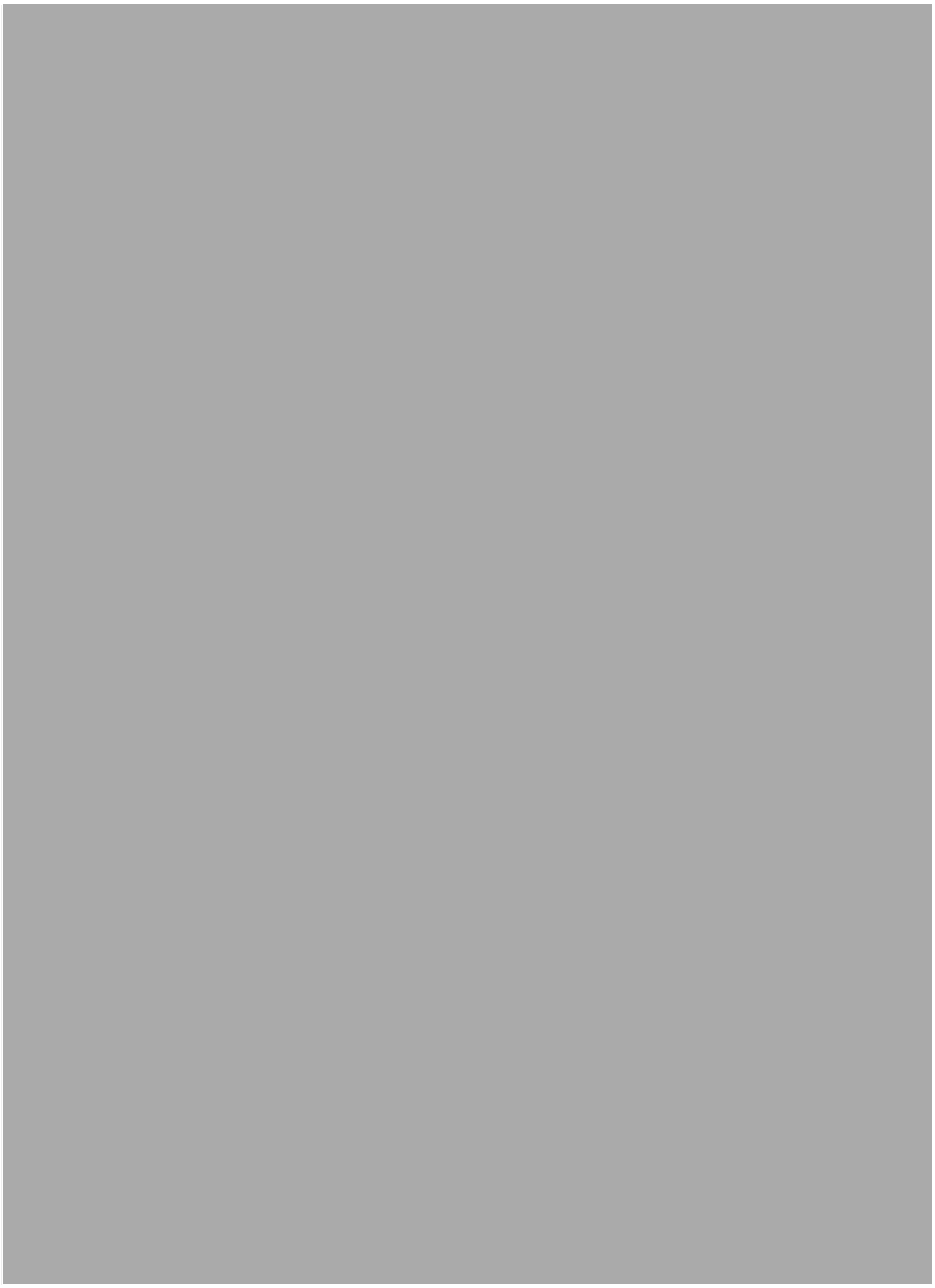
허가를 위해 필요한 많은 실험자료를 확보하고 있음

- 고속로설계부와 ANL이 상호 협력에 대하여 지속적으로 논의하여 2012년 5월 2~4일 ANL 전문가 4인이 KAERI를 방문하여 기술 토의를 진행한 바 있음
  · 현재는 사업단이 ANL 협력 업무를 연구부서로부터 이관 받아 수행 중
- 2012년도 협력 업무는 SFR 원형로 개념설계 리뷰, 특정설계 기술개발과 인허가 획득을 위한 필요 업무 도출 등이었으며, 2013년도 협력 업무는 다음과 같음
  · FY12년 업무는 원형로 개념설계 보고서 검토 및 Trade-off 연구에 집중하여 원형로 개념을 적기에 결정하도록 주력하였음
  · FY13년 업무는 원형로 설계를 위한 각 계통/기기의 설계요건을 설정하는 역할을 담당하고 설계에 부분적으로 참여할 것임
  · ANL의 업무는 크게 기술검토, 설계참여 및 검증자료 확보의 3가지로 구분됨
- 2012년도 협력의 소요 예산은 약20억원(US$1,700,000.00)이었으며, 2013년도 협력의 소요 예산은 약 25억원(US$2,080,000.00)임
  · 매 회계년도의 업무와 예산은 전년도에 구체적으로 협의하여 결정함
- KAERI와 ANL의 구체적 업무분장, 사용 전산코드 및 방법론 결정, 상세일정 작성은 매 회계연도 시작 전에 구체적으로 논의할 것임
  · 전산코드 및 실험자료 등의 Applied Technology(AT)는 DOE의 허가를 받아야 하는데, 인허가 획득시까지 필요한 자료들은 지속적으로 정리하여 확보할 것임
  · ANL에서 전산코드는 별 문제가 없다고 판단하고 있으며, 금속연료 조사시험 자료와 안전 관련 실험자료는, KINS에서 요구하는 자료에 대한 논의를 바탕으로 DOE에 수시로 경과보고하며 필요한 자료들의 허가를 받는 과정이 필요함
- 향후계획
  · 협력 내용을 추진하기 위해 다수의 KAERI의 연구진이 ANL을 방문하여 공동 작업을 진행할 것이며, 정기적으로 양측을 오가며 정기적으로 진행 상황을 점검하기로 하였음
  · ANL과의 협력 논의와 계약 체결은 사업단 책임 하에 추진할 것임

표 3 KAERI-ANL WFO Agreement (No. 85E20)

Page 1 of 11





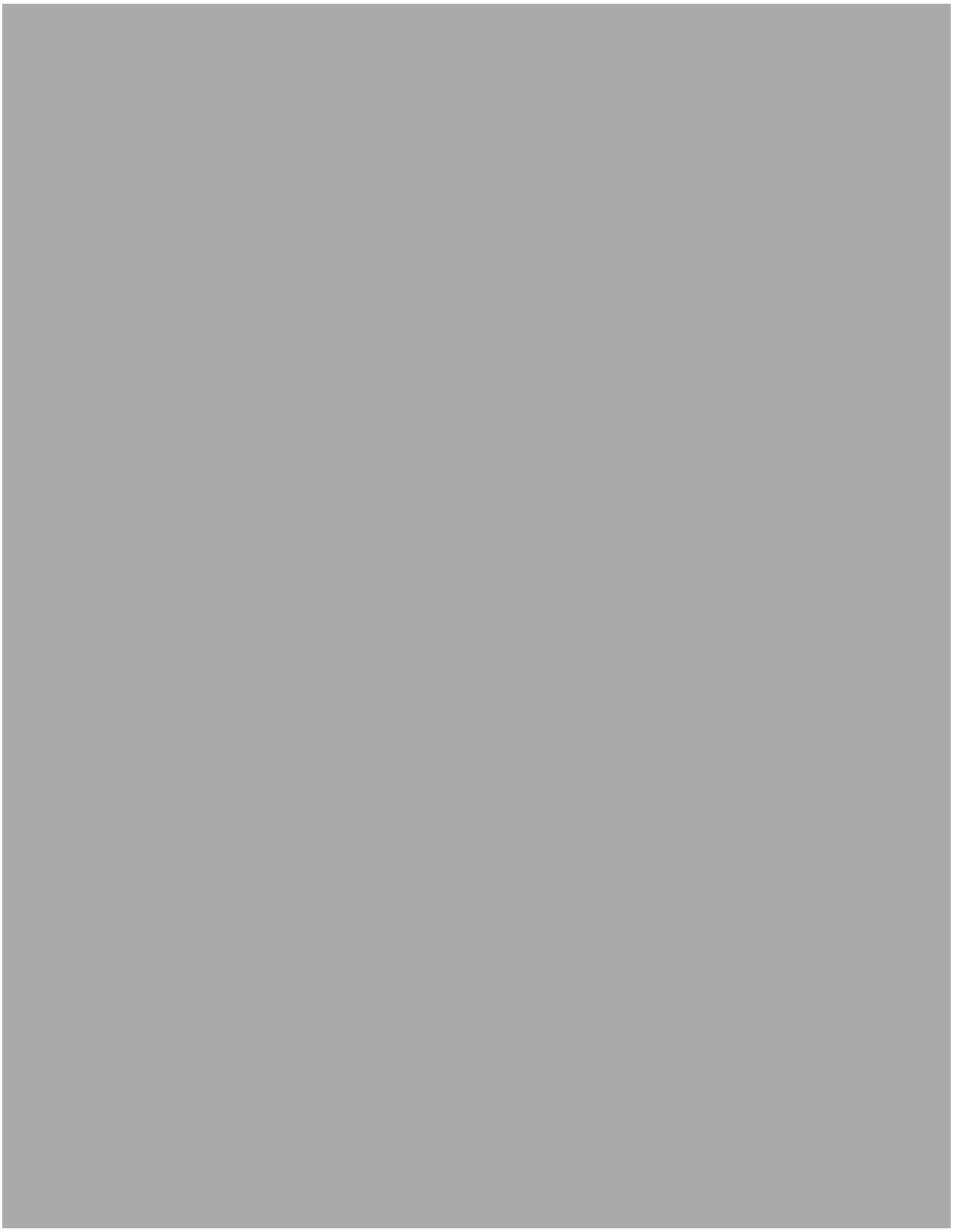


표 4 ANL 업무기술서(Statement of Work) (FY12)

APPENDIX A

P-12021
KAERI-ANL Joint Program on Design Development of a Prototype
Sodium-cooled Fast Reactor

Work proposed by:

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July 2012

Statement of Work

KAERI-ANL Joint Program on Design Development of a Prototype Sodium-cooled Fast Reactor

1. Purpose

KAERI has been developing the KALIMER sodium-cooled fast reactor concept and recently decided to develop a 100 MWe size prototype in order to reduce the construction cost at the same time maintaining the design features desirable for commercialization. The most important innovation in the KAERI SFR is the utilization of metal fuel which enables unique inherent safety characteristics as well as a most efficient actinide burner.

The underlying key technology for the KAERI SFR is based on the fast reactor technologies developed at Argonne National Laboratory (ANL) in the 1980s and 90s. Therefore, a joint program between KAERI and ANL to develop a prototype SFR will be mutually beneficial. The joint program will be performed under a Work-for-Others Contract for the ANL portion of the work.

The purpose of this Statement of Work is to describe the ANL tasks in support of the joint program to develop a Specific Design in order to secure a Design Approval by the Korean licensing authority. This joint program will also lay the groundwork for a potential future licensing of a sodium-cooled fast reactor by the U.S. Nuclear Regulatory Commission.

KAERI will be responsible for the overall SFR project and will have a much bigger role in all aspects of the design development and the licensing support activities than outlined here for the ANL portion. The ANL contributions will be fully integrated into the KAERI activities.

2. Task Description

Task 1 Specific Design of KAERI SFR

This task includes all design related activities associated with reactor core, reactor enclosure system, primary heat transport system, intermediate heat transport system, shutdown heat removal system, fuel handling system, instrumentation and control system, in-service inspection system, power conversion system, and buildings and structures as required to develop the Specific Design.

The design activities under this task will integrate key design features from KALIMER and past Argonne SFR designs and incorporate the lessons learned from the SFR operations worldwide.

Subtask 1.1 System Functional Requirements and Overall Process Flow Diagram

The system functional requirements will be developed in order to guide the structures, systems and components design activities. In addition, an overall process flow diagram will be developed for the reactor system, primary and intermediate heat transport systems, and main steam and feedwater systems. These system functional requirements and overall process flow diagram will be updated on an annual basis to take account of the design progression.

Subtask 1.2 Reactor Core Design

An optimized reactor core design specification that best meets the design goals will be developed through detailed design tradeoff studies involving the pin design parameters, burnup capability, fluence limits, thermal hydraulics, excess reactivity control, fueling interval, and other considerations. This subtask requires a close coordination with KAERI design activities in the early stages in order to establish a reference core design specification, which will form the basis for other design activities to proceed and also interface with the safety analysis activities. The activities under this subtask include detailed analyses and designs for the following:

- Tradeoff studies on power level and core design alternatives
- Reference core design determination
    - Reactivity control system
    - Steady-state thermal-hydraulics analysis
    - Kinetics parameters and reactivity coefficients as required for the safety analyses
    - Shielding
    - Fuel system design

Subtask 1.3 Reactor Enclosure System

Based upon the reference core envelope developed in Subtask 1.2, the reactor enclosure system design parameters and configurations will be developed. The design and engineering analyses will include the following major structures, systems, and components:

- Reactor vessel and support structure
- Guard vessel and support structure
- Reactor vessel head

All engineering analyses will be performed at nominal steady state conditions for anticipated normal loadings.

Subtask 1.4 Primary Heat Transport System

Based upon the reference core envelope developed in Subtask 1.2, the primary heat transport system design parameters and configurations will be developed. The design and engineering analyses will include the following major structures, systems, and components:

- Primary pump – conceptual specifications
- Intermediate heat exchanger – conceptual specifications
- Internal piping – sizing and design layout
- Lower internal structure – sizing and design layout
- Upper internal structure – conceptual specifications
- Core barrel
- Core restraint structure
- Redan

All engineering analyses will be performed at nominal steady state conditions for anticipated normal loadings. For the primary pump and intermediate heat exchanger, the work will include conceptual specifications including a thermal-hydraulic analysis to verify component sizing. Multi-dimensional thermal hydraulic analyses of reactor pool will be performed in conjunction with Task 2.

Subtask 1.5 Intermediate Heat Transport System

Consistent with the system functional requirements that are developed under Subtask 1.1, the intermediate heat transport system (IHTS) design parameters and configurations will be developed. This subtask will include development of:

- Overall piping specifications, sizing, and design layout, including expansion tanks for each loop,
- Steam generator conceptual specifications including thermal analysis to verify component sizing for the overall plant thermal load,
- Design and specifications for the Sodium-Water Reaction Pressure Relief System (SWRPRS) that safely depressurizes and drains the IHTS during a postulated steam generator tube rupture event,
- Sizing and specifications for the IHTS sodium pumps; both mechanical and electromagnetic (EM) pump designs will be considered as part of the task,
- Sodium purification (cold trap) design and specifications including plugging meters for quantitative evaluation of sodium purity level, and
- IHTS instrumentation layout and specifications.

Once the base design is developed, various analyses will be carried out to verify proper functionality under nominal steady-state conditions. Structural and thermal stress analyses will be carried out with ANSYS to verify acceptable stress levels in the system for the design lifetime of the plant. In addition, the structural integrity of the IHTS under a postulated large steam generator tube rupture event will be verified with the SWAAM sodium-water reaction code. This analysis will also provide the peak design pressure for the intermediate heat exchanger (IHX) design that will be developed under Subtask 1.4.

Subtask 1.6 Shutdown Heat Removal System

Consistent with the system functional requirements developed in Subtask 1.1, the shutdown heat removal system will be designed and analyzed. The preliminary specific design will include a trade study of the various options for decay heat removal and their impact upon reactor performance and reliability. Based upon this initial trade study, a SHRS approach will be decided and a preliminary specific design with appropriate engineering analyses performed that demonstrates the feasibility of the design under natural circulation conditions (loss of electrical power) and nominal steady state conditions. A description of the SHRS functions and requirements will also be prepared.

- Development of design concepts of shutdown heat removal system
- Evaluation of system and component performances

Subtask 1.7 Fuel Handling System

Based upon the reference core assembly size, the in-vessel and ex-vessel fuel handling mechanisms will be designed. The fuel handling system will include the following:

- In-vessel fuel handling mechanism and control system
- Rotatable plugs
- Ex-vessel fuel handling mechanism – physical size estimates, functions and requirements, and conceptual specifications
- Transfer cask technology – physical size estimates, functions and requirements, and conceptual specifications

Once the base design is developed, various engineering analyses will be performed at nominal steady-state conditions to verify proper functionality. Shielding requirements will also be determined for those systems located outside of the reactor vessel.

Subtask 1.8 Instrumentation and Control System

This work will focus on evaluating the various possibilities for making optimum use of the passive features and the inherent safety response in the design of the I&C system. The objective is to minimize safety channels and instrumentation in the plant protection system (PPS) and to utilize passive features and inherent feedbacks in the plant control system (PCS). The design activities will include:

- Steam generator leak detection and non-destructive examination techniques,
- Failed fuel detection and location system,
- Non-invasive instrumentation to minimize the number of instrumentation penetrations, thimbles and cabling,
- Component condition monitoring and diagnostics capabilities,
- Inherent response-based algorithms for plant controllers,
- PPS/PCS simulation to develop controller algorithms, instrument selection and location, and trip points.

- Instrumentation layout and specifications for the primary heat transport system, intermediate heat transport system, and shutdown heat removal system, if resources become available.

Subtask 1.9 In-Service Inspection

Innovations in inspection and maintenance technologies, e.g. under-sodium viewing, mechanical indexing and remote handling, will be evaluated for in-service inspection application. Innovative concepts may include ultrasonic sensor scanning for under-sodium viewing, safety measures in fuel handling, and partially automated remote handling and inspection devices. In order to enable the assessment of plant inspectibility and maintainability, a simulation environment will be developed, which allows test operation of in-service inspection and maintenance technologies in virtual environment. Integrating recent innovations in multi-modal display, distributed computer operating systems, and remote systems technologies and nuclear simulation technologies will allow accurate simulation to be conveyed in realistic multi-modal experience.

Subtask 1.10 Power Conversion System

The overall process flow diagram developed under Subtask 1.1 will include the power conversion system. Additional activities, such as plant operation strategy and control logics and plant performance analysis will be performed as the need arises.

Subtask 1.11 Buildings and Structures

The performance of the reactor building with and without a seismic isolation system will be analyzed. If it is decided to incorporate a seismic isolation system, then appropriate seismic isolation technologies for the reactor building will be recommended and incorporated into the reactor building design.

- Plant layout based on the reactor and fuel handling, and turbine building design
- Application of seismic isolation system for the reactor building and other structures as necessary

Task 2 Safety Analyses

This task consists of three main activities: development of the safety design criteria, safety analyses to be conducted in coordination with the design activities under Task 1, and preparation of safety documents to be submitted to the licensing authority.

Subtask 2.1 Establishment of Safety Design Criteria

The USNRC's General Design Criteria for Nuclear Power Plants (10 CFR Part 50 Appendix A) dictate the safety design and licensing of LWRs. There are no equivalent criteria for SFRs. Therefore, it is essential for the prototype SFR Project to take the initiative to develop and propose the safety design criteria, which will guide the safety design approaches as well as the licensing processes. The proposed safety criteria should be robust so that they form the foundation for the design activities, as well as being accepted by the licensing authority.

The ANSI Standards on SFR Safety Design Criteria and the NRC licensing reviews for the FFTF, CRBR, and PRISM Projects will be evaluated to recommend the Safety Design Criteria applicable for this project. The on-going activities of the ANS Standards Committee and the GIF Task Force will also be factored in. In particular, potential application of the risk-informed and/or performance-based regulatory approach will be evaluated in developing the Safety Design Criteria.

Consistent with the Safety Design Criteria, the next tier design-basis-events and beyond-design-basis-events will be developed, which need to be analyzed in detail in Subtask 2.2. DBEs and BDBEs adopted in the FFTF, CRBR, and PRISM Projects will form the base, however, a new set of DBEs and BDBEs will be developed considering the evolutions in design and safety approach.

Subtask 2.2 Safety Analyses

The design-basis-events and beyond-design-basis-events developed in Subtask 2.1 will be analyzed using the Argonne developed SAS4A and SASSYS codes and other codes as necessary.

The bounding events will be identified that will establish the source term and the containment design basis.

In order to assure that the desired inherent safety features are built in to (not added on to) the design, Task 2 will be conducted in close coordination and consort with Task 1 so that the safety criteria are met though design modifications. Based on the trade studies which were completed years 1-2, the design options will have been selected and a baseline design will have been established. As details and refinements of this baseline design are developed to support the specific design, additional more detailed safety analyses will be pursued in years 2-4 to provide input to the PSID.

The analyses will cover the entire range of operating transients including normal power maneuvering, anticipated operation sequences, design-basis events, as well as beyond-design-basis events. As needed, the simulations will also cover the whole-plant dynamics including the detailed steam-generator model and energy conversion system response. The list of the transients (duty cycles) will include sufficient details to facilitate the modeling of the events with a dynamics code as well as an anticipated frequency of occurrence for each event. The list for transient analysis will be prioritized based on the PSA results from Subtask 2.3.

The results of the normal operational transients and design basis events will include consideration of uncertainties in the analysis to investigate the dynamics characteristics and performance of the entire plant, including the reactor and BOP sides. Uncertainties in the safety case evaluations and loading histories for the component design will be incorporated in the analyses either via utilization of hot channel factors or by employing an uncertainty quantification scheme based on sampling of input variables. Based on the results of the transients, the feedback on the reactor, BOP and control system design and functions will be provided. In addition, the results of the transients will be provided for duty cycle structural analyses of various components, such as the reactor vessel.

- Transient analysis for anticipated operation sequences (reactivity insertion, loss of core cooling, loss of normal heat sink)
- Analysis for design basis events (LOF, TOP, LOHS, partial flow blockage, pipe break, main vessel leak, sodium leakage, SG tube rupture, etc.)
- Analysis for beyond-design basis events (UTOP, ULOF, ULOHS, flow blockage, large SG tube rupture, sodium leakage)

Subtask 2.3 Probabilistic Safety Assessment (PSA)

Another important activity under the safety analyses is associated with the probabilistic safety assessment (PSA). An appropriate methodology for PSA will be developed in the early phase and Level-1 and Level-2 quantification will be followed. Using preliminary design features, an initial model can be developed and analyzed to identify which accident sequences could be candidates for more detailed study and analysis. The model can be fine-tuned and expanded to include more event types (such as external events) with more site-specific details.

- Finalization of PSA methodology
- Establishment of the basis for mechanistic source term
- Level-1 PSA
- Level-2 PSA

Subtask 2.4 Preliminary Safety Information Document (PSID)

As sufficient results have been obtained from the safety analyses activities under subtask 2.2 and design iterations have been completed with the Task 1 activities, the preliminary safety information document (PSID) will be prepared. The purpose of the PSID is to submit it to the Korean licensing authority as the basis for a preliminary determination of licensability and assessment of safety issues in the design.

Subtask 2.5 Preliminary Safety Analysis Report (PSAR)
The preliminary safety analysis report (PSAR) will be developed incorporating the findings in the Safety Evaluation Report for the PSID.

Task 3 Licensing Support

Once the PSID is submitted to the licensing authority, there certainly will be technical questions raised, which would require clarifications or additional analyses. This task will provide the technical support during the licensing review process. More importantly, the computer codes and the database used in the safety analyses will be verified and validated as described in the following subtasks.

Subtask 3.1 Validation of Neutronics Design Computer Codes

The neutronics methodologies, including multi-group cross section generation, whole-core neutronics calculations and depletion calculations, will be validated against appropriate benchmark problems and/or appropriate critical experiments data.

It is assumed that the physics database used for the validation of ANL neutronics codes will be available to Korea Institute of Nuclear Safety upon request during the licensing review of corresponding topical reports.

Subtask 3.2 Validation of Fuel Design Basis

This subtask will document the U.S. experience with metallic fuel (U-Zr and U-TRU-Zr) and fuel components (cladding and duct), in particular its irradiation performance in EBR-II, FFTF, and TREAT, including steady state, transient and run-beyond cladding breach performance.

Existing models for important phenomena related to fuel performance such as FCCI, constituent redistribution, fission gas induced swelling, thermo-mechanical models, etc., will be evaluated and in collaboration with KAERI an advanced fuel performance code will be developed and validated against the EBR-II and FFTF experimental database.

In addition, existing out-pile experimental data will be evaluated and will assist KAERI in planning execution of additional out-of-pile experiments.

Subtask 3.3 Validation of Safety Analyses Methodology

The SAS4A/SASSYS-1 code system has been used extensively as a design basis analysis tool for the EBR-II, FFTF and CRBR reactors, as a conceptual design evaluation code in the U.S. DOE reactor development projects (e.g. LSPB, SAFR, PRISM) as well as in severe accident calculations for the FFTF and CRBRP reactors in the U.S., for the SNR-300 reactor in Germany, for the MONJU reactor in Japan, for the PEC reactor in Italy, and for the BN-600 reactor in Russia. The models in SAS4A/SASSYS-1 have been validated with extensive analyses of laboratory experiments and in-pile fuel tests in the TREAT and CABRI as well as with reactor and plant test data from EBR-II and FFTF. An updated compilation of the past verification and validation reports of SAS4A/SASSYS-1 will be prepared.

These international validation bases will be extended to support the design and licensing of the KAERI SFR. Since the SAS4A/SASSYS-1 code system has been developed primarily as a design analysis tool, using it as a licensing tool would require extensive effort for compliance of software quality assurance practices required by national licensing authorities. Therefore, bulk of the task will be devoted to collaboration with

KAERI staff to place the SAS4A/SASSYS-1 code system under configuration control, to establish a vigorous code verification and QA plan for code maintenance, and testing of software components through improved software quality engineering practices.

The creation of an input and post processor as an improved user interface to reduce potential input errors can also be performed under this task in collaboration with KAERI analysts. Further improvements to support parallel applications by taking advantage of standard parallel computing platforms can allow utilization of the SAS4A/SASSYS-1 code system as the simulation engine for the automated design optimizations, as well as the uncertainty quantification and sensitivity analysis schemes. If the design of the KAERI SFR is to withstand the regulatory scrutiny especially in at the specific design stage, the software system that supports the license application will likely be required to have these capabilities in place.

3. Computer Codes to be used for the ANL Tasks

There are 4 categories of computer codes that Argonne personnel plan to use in the execution of the ANL tasks:

1. ANL codes available (or planned to be) from RSICC or NEA Data Bank
   - ETOE/MCC-3 (update of MCC-2)
   - DIF3D
   - REBUS-3
   - SWAMM
2. ANL codes not available from RSICC or NEA Data Bank – DOE's "Applied Technology"
   - SAS4A/SASSYS-1
3. ANL codes that will be further developed during this project
   - VARI3D
   - LIFE-METAL
   - NUBOW-3D
4. Commercial codes with ANL interface routines
   - SUPERENERGY-2-ANL
   - STAR-CD/STAR-CCM+
   - ANSYS
   - SAP2000

KAERI already has access to the computer codes in Category 1 above.

If KAERI desires to have access to any computer code in Category 2, an explicit written approval by DOE-NE is required since they are categorized as DOE's "Applied Technology." ANL will be responsible to secure the DOE approval.

The computer codes in Category 3 have not been fully developed for release. However these codes will play an important role for the ANL tasks, and hence the development will be resumed for ultimate use for the project. When these codes are ready for release and if KAERI desires to have access, the same procedure discussed above for Category 2 will apply here as well. KAERI is welcome to participate in these code development activities, at KAERI's expense, however the final coding changes and documentation will be the responsibility of ANL in order to maintain the necessary quality assurance.

If KAERI wishes to use any of the Category 4 computer codes, it is KAERI's responsibility to acquire the license for these codes. ANL will share the interface routines and experiences.

4. Project Schedule and Milestones

The project schedule and major milestones are as follows:

- Project Duration: March 1, 2012 through February 28, 2021
- Complete Conceptual Design: February 28, 2013
- Complete Preliminary Safety Information Document: February 28, 2016
- Complete Preliminary Design: February 28, 2016
- Complete Safety Analysis Report: February 28, 2018
- Complete Specific Design: February 28, 2018
- Complete Design Approval by Licensing Authority: February 28, 2021

5. Project Efforts and Costs

The Project efforts (in terms of man-months) are summarized in the table below in subtask levels for the first 5 years.

	Year 1	Year 2	Year 3	Year 4	Year 5
1. Specific Design					
1.1 Requirements	2	4	2	2	2
1.2 Reactor Core	8	20	14	6	6
1.3 Enclosure	6	16	16	14	13
1.4 PHTS	10	24	24	22	22
1.5 IHTS	9	21	21	20	19
1.6 SHRS	6	15	15	14	12
1.7 Fuel Handling	9	21	21	20	18
1.8 I&C	2	6	6	6	6
1.9 ISI	1	4	4	4	4
1.10 PCS					
1.11 Buildings	2	6	6	6	6
2. Safety Analyses					
2.1 Design Criteria	2	4	2	2	2
2.2 Safety Analyses	4	22	22	24	24
2.3 PSA	1	13	13	17	20

2.4 PSID		13	14	24	
2.5 PSAR					24
3. Licensing Support					
3.1 Neutronics	2	7	13	13	13
3.2 Fuel Basis	2	7	13	13	13
3.3 Safety	2	7	13	13	13
Total (man-month)	68	210	219	220	217

The total project costs (in thousand dollars) are summarized in the table below.

	Year 1	Year 2	Year 3	Year 4	Year 5
Salaries	992	3,022	3,215	3,306	3,328
Fringe Benefits	199	788	821	864	854
Indirect Costs	409	1,490	1,564	1,630	1,618
M&S, Travel	100	200	200	200	200
Total	**$1,700**	**$5,500**	**$5,800**	**$6,000**	**$6,000**
	Year 6	**Year 7**	**Year 8**	**Year 9**	**Total**
Total Salaries	3,289	3,226	3,169	3,169	26,716
Fringe Benefits	864	903	936	936	7,165
Indirect Costs	1,647	1,671	1,695	1,695	13,419
M&S, Travel	200	200	200	200	1,900
Total	**$6,000**	**$6,000**	**$6,000**	**$6,000**	**$49,000**

The annual budget presented above is subject to change based on the detailed work plan to be developed for each year and the funding availability. ANL will develop, in consultation with KAERI, the detailed work plan for each year by January 1. The annual budget will be adjusted as necessary by mutual agreement and the WFO Agreement will be amended as necessary.

6. Project Meetings

The joint project meetings will be scheduled on a regular basis (between 2 to 3 months) alternating the meeting places between KAERI and ANL to review the technical progress and to decide on the future action items.

7. KAERI Assignees at ANL

For the purpose of joint participation in the project and to fulfill the liaison role, a mutually agreed upon number of KAERI staff can be assigned to stay at ANL for the duration of one year for each assignee. KAERI will be responsible for the entire cost of assignment including salaries, travel and living costs, and health and other insurance costs. ANL will provide the office facilities with desktop computers and supplies.

8. Project Deliverables

The deliverables will include:

- ANL contributions to Functional Requirements and Specifications Document
- ANL contributions to Conceptual Design Report
- ANL contributions to Preliminary Safety Information Document
- ANL contributions to Preliminary Design Report
- ANL contributions to Safety Analysis Report
- ANL contributions to Specific Design Report
- Computer codes as agreed to in Section 3.

In addition, other deliverables can be defined at the regularly scheduled project meetings based on mutual agreements.

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Supplement to
APPENDIX A
Statement of Work

Consistent with the Statement of Work, this Supplement is prepared to elaborate on more details of the Argonne activities to be performed under the WFO Agreement.

1. Relationship between the work performed at ANL under this Statement of Work and the activities carried out at KAERI

The Statement of Work attached to the Work for Others (WFO) Agreement describes the activities performed by Argonne National Laboratory in support of the KAERI's Prototype Sodium-cooled Fast Reactor (SFR) Project. Therefore, the KAERI's activities are not included in this document. However, the KAERI's SFR Project will be a joint effort between KAERI and ANL for the workscope described in the Statement of Work. The design will be developed jointly; the safety analyses and documents submitted to the licensing authority will be jointly developed; and the defense in licensing will also be a joint effort. However, KAERI will be responsible for the overall SFR project and will have a much bigger role in all aspects of the design development and the licensing support activities than outlined in Appendix A for the ANL portion. The ANL contributions will be fully integrated into the KAERI activities. Further, KAERI shall have unlimited rights (right to use, duplicate or disclose, and permit others to do so) to ALL technical data produced by ANL in the performance of the work under this agreement.

The most important factor for the ANL's participation in this joint effort is due to the ANL's unique qualifications. Argonne's qualifications are unique in two respects. First, ANL designed, constructed, and operated Experimental Breeder Reactor-I (EBR-I) and EBR-II. Furthermore, ANL was extensively involved in the Clinch River Breeder Reactor (CRBR), Fast Flux Test Facility (FFTF), Large Scale Prototype Breeder (LSPB), Large Pool Plant (LPP), Sodium Advanced Fast Reactor (SAFR), and Power Reactor Innovative Small Module (PRISM) projects both in design and licensing support activities. Recently ANL also carried out the Small Modular Fast Reactor (SMFR) in collaboration with Japan Atomic Energy Agency and French Atomic Energy Commission and Advanced Burner Test Reactor (ABTR) in support of the Global Nuclear Energy Partnership (GNEP) initiative. ANL will incorporate the lessons learned from these numerous SFR projects into the KAERI SFR Project to make it a model prototype. The SFR experience worldwide has a mixed record of operation. Understanding what has worked, what has not worked, and why and applying them from the onset of the conceptual design is crucial. Argonne designers bring such expertise.

In addition, Argonne's experience in licensing of the past SFRs led to its conviction that for the future SFRs to be viable, inherent passive safety has to be exploited with emphasis on prevention of severe accidents than the mitigation features and myriad of engineered safety systems. Argonne will support KAERI to bring about a new paradigm in licensing approach for this SFR Project.

The other aspect of Argonne's unique qualifications is associated with its R&D accomplishments during the Integral Fast Reactor (IFR) program in the 1980s and 90s. The R&D investments during these periods exceed $1 billion, including the operating costs of the supporting facilities. The metal fuel development was indeed the foundation of the IFR program. The extensive database developed by Argonne on the metal fuel irradiation performance and its safety characteristics would be essential for the preparation of the safety documents to be submitted to the licensing authority and for a successful design approval process. The creation of the equivalent fuels and safety database by KAERI alone would require more than a decade of concentrated R&D efforts and extensive resources.

2. Detailed Plan for ANL Activities in FY2012

The main objective of FY2012 activities is to complete a conceptual design, which will form a reference design and starting point for: (1) further optimization and design tradeoffs of each system and subsystems, (2) initiation of integrated safety analyses, and (3) evaluation of the system thermal-hydraulic and transient analyses.

Therefore, about 80% of the Argonne efforts will be focused on the Task 1 conceptual design.

Task 1 Specific Design of KAERI SFR (55mm)

Subtask 1.1 System Functional Requirements and Overall Process Flow Diagram (2mm)

KAERI will develop the fast reactor system functional requirements that will guide the structures, systems and components preliminary specific design activities, and ANL will review and provide comments. This task will also include the development of an overall process flow diagram for the reactor system, the primary and intermediate heat transport systems, and main steam and feedwater systems. An initial set of functional requirements will be developed in this period and will be updated on an annual basis to take into account for the design progression.

Subtask 1.2 Reactor Core Design (8mm)

Establishing a well optimized core design is crucial for the success of the overall reactor design because the design of the rest of the reactor enclosure system, fuel handling system and so on depends on the core envelope and the safety analyses cannot be initiated without the reference core specifications.

An optimized reactor core design specification that best meets the design goals will be developed through detailed design tradeoff studies involving the pin design parameters, burnup capability, fluence limits, thermal hydraulics, excess reactivity control, fueling interval, and other considerations. This subtask requires a close coordination with KAERI design activities in the early stages in order to establish a reference core design specification, which will form the basis for other design activities to proceed and also interface with the safety analysis activities.

Subtask 1.3 Reactor Enclosure System (6mm)

Based upon the reference core envelope developed in Subtask 1.2, the reactor enclosure system design parameters and configurations will be developed to the conceptual level. This first year work will include the design of the reactor vessel and its support structure, the guard vessel and its supporting structure and the reactor vessel head. The design concept and initial supporting engineering analyses will include the following major structures, systems, and components:

- Reactor vessel and support structure – this includes the vessel that holds the primary sodium and all of the reactor internals. The support structure supports the reactor vessel and connects the reactor vessel to the reactor building.
- Guard vessel and support structure – this includes the vessel that surrounds the reactor vessel and is used to collect any sodium that may leak from the reactor vessel. The support structure for the guard vessel will be design that connects the guard vessel to the reactor building. Sufficient room will be maintained between the reactor vessel and guard vessel to provide for in-service inspection.
- Reactor vessel head – this activity includes the head of the reactor vessel that provides support for the various systems and components that penetrate through the reactor vessel such as the primary pumps, intermediate heat exchangers, and rotatable plugs.

Any initial engineering analyses will be performed at nominal steady state conditions for anticipated normal loadings.

Subtask 1.4 Primary Heat Transport System (10mm)

Based upon the reference core envelope developed in Subtask 1.2, the primary heat transport system design parameters and configurations will be developed to the conceptual level. The design and engineering analyses will include the following major structures, systems, and components:

- Primary pump – the primary pump will be sized to the conceptual level to ensure that sufficient space will be available within the primary reactor vessel for this component. The primary pump provides the main coolant flow through the reactor core and intermediate heat exchanger.
- Intermediate heat exchanger – the intermediate heat exchanger will be sized to the conceptual level to ensure that sufficient space will be available within the primary reactor vessel for this component. The intermediate heat exchanger is a sodium-to-sodium heat exchanger that provides for the isolation between the primary radioactive coolant and the high pressure balance of plant fluid.

- Internal piping – sizing and design layout to the conceptual level. This internal piping will be located between the primary pumps and the inlet plenum.
- Lower internal structure – sizing and design layout of the upper and lower grid plate structure will be performed along with the space between the two plates.
- Upper internal structure – high level requirements for the upper internal structure will be prepared. Space will be left within the reactor vessel to accommodate the design of the upper internal structure.
- Core barrel – Based upon the reactor core design, the core barrel will be sized to the conceptual level.
- Core restraint structure – Based upon the reactor core design, initial sizing and location of the core restraint ring and load pads will be performed.

For the first year, any initial engineering analyses will be performed at nominal steady state conditions for anticipated normal loadings. For the primary pump and intermediate heat exchanger, the work will include preliminary thermal-hydraulic analysis to verify component sizing.

In addition, conceptual layout of failed fuel detection and location system will be developed in collaboration with the Task 1.2.

Subtask 1.5 Intermediate Heat Transport System (9mm)

Consistent with the system functional requirements that are developed under Subtask 1.1, the intermediate heat transport system (IHTS) design parameters and configurations will be developed to the concept design level. This subtask will include development of:

- Overall piping sizing and design layout, including expansion tanks for each loop,
- Steam generator thermal analysis to verify rough component sizing for the overall plant thermal load including nominal height, diameter, tube count, and tube sizing,
- Sizing and specifications for the IHTS sodium pumps; both mechanical and electromagnetic (EM) pump designs will be considered as part of the task,
- Sodium purification (cold trap) sizing will be performed based upon the volume of the sodium in the intermediate heat transport system
- Specifications for the sodium-water reaction pressure relief system.

Subtask 1.6 Shutdown Heat Removal System (6mm)

Consistent with the system functional requirements developed in Subtask 1.1, a concept for the shutdown heat removal system will be developed. The conceptual design will include a high level comparison study of the various decay heat removal options. Based upon this initial comparison study, a SHRS approach will be recommended and the SHRS design will be sized and incorporated into the design model.

- Development of design concepts for shutdown heat removal system
- Evaluation of system and component performance

Subtask 1.7 Fuel Handling System (9mm)

Based upon the reference core assembly size, a concept for the in-vessel and ex-vessel fuel handling mechanisms will be created during the first year. The fuel handling system will include the following system and components:

- In-vessel fuel handling mechanism that is used to move the spent fuel from the core region to a suitable transfer location
- Rotatable plugs that are used for providing placement for the in-vessel fuel handling mechanism over the appropriate reactor core location
- Ex-vessel fuel handling mechanism – physical size estimates, functions, and conceptual specifications. The ex-vessel fuel handling mechanism is used to transfer spent fuel from the reactor vessel to the transfer cask
- Transfer cask conceptual specification

Once the base design is developed, various engineering analyses will be performed at nominal steady-state conditions to verify proper functionality. Shielding requirements will also be determined for those systems located outside of the reactor vessel.

Subtask 1.8 Instrumentation and Control System (2mm)

For the first year, this work activity will create a list of the necessary I&C systems and develop an I&C strategy for the primary and intermediate heat transport systems. In addition, a specific task on steam generator leak detection system will be initiated.

Subtask 1.9 In-Service Inspection (1mm)

For the first year of this project, an in-service inspection requirements chart for the primary plant will be prepared. This chart will include information on required primary plant in-service inspections and technology options for performing those inspections.

Subtask 1.10 Power Conversion System

No specific design activities are defined at this time pending further discussions with KAERI. However, the overall process flow diagram developed under Subtask 1.1 will include the power conversion system.

Subtask 1.11 Buildings and Structures (2mm)

For the first year, the reactor building will be sized to accommodate the primary and intermediate heat transport system equipment developed in the above tasks.

Task 2 Safety Analyses (7mm)

The first year safety tasks will focus on the investigation of the design extension conditions (DECs, also known as the beyond design basis accidents), such as the double-fault events during which the reactor scram system is assumed to fail. Although probability of double-fault events is very low, the reactor will be designed to withstand such accidents without core damage. Therefore, goal of the first year safety analysis will be to show that radioactivity release to the environment can be prevented even during the most unlikely double fault events, and to ensure that the safety is built-in (as opposed to added-on) to the design. The analyses of design extension conditions will be done on best-estimate basis (no extra uncertainty margins, such as hot channel factors, will be included).

A complete safety analysis envelope during the specific design stage will also include consideration of more likely anticipated operational occurrences (AOOs) and design basis events (DBEs). Those analyses require more detailed knowledge of the reactor and the balance-of-plant design (which are expected to be available at the end of the first year as the design matures beyond conceptual design stage), and they involve uncertainties. Therefore, the analyses of AOOs and DBEs with uncertainties are expected to begin after the first year.

Subtask 2.1 Establishment of Safety Design Criteria (2mm)

A set of safety design criteria will be prepared in coordination with KAERI as the basis for common safety approach for design and safety assessment. The commonly used safety criteria in SFR designs include specification of key safety related components and standard safety analysis practices such as avoiding:

- sodium coolant boiling in the core at all times and at all locations,
- fuel melting, and
- cladding failure due to various mechanisms, including the fuel-cladding mechanical and/or chemical interactions.

Typically, the safety design criteria specify sufficient margin in compliance with the national regulatory requirements. Parallels with the ongoing efforts under the Generation-IV SFR Safety Design Criteria Task Force and ANS 54.1 standards will be drawn as the fundamental approach to establishment of common safety design criteria.

Subtask 2.2 Safety Analyses (4mm)

The most challenging accidents are so-called unprotected events where the second fault includes the failure of the reactor protection system to shut down the fission reaction in the core. The first fault can be any event that introduces significant misbalance between the heat produced by the reactor and the ability of the cooling system to remove this heat from the core. The past experience has shown that the most severe results are expected for the unprotected event of the primary coolant pump failure (so-called unprotected loss-of-flow, or ULOF, event), unprotected loss of heat sink (ULOHS) event, or unprotected control rod ejection (unprotected transient over power, or UTOP, event). Those anticipated transients without scram have been historically the focus of the SFR safety analysis, and they will also be given the priority for the safety analysis during the first year.

The first year's work will also focus on evaluating the various possibilities for making optimum use of the passive features and the inherent safety response in the safety case for the design. Parametric trade studies will be performed in conjunction with the other design tasks to down select the design options and ensure that safety is built into the design, not added onto. Iterations will be carried out with the design subtasks in the down-selection process to define options.

The safety analyses conducted during the first year will provide input for the PSID in following years. Anticipated operational transients and design basis events will be established to provide the initial strategy for the spectrum of events to be considered in the PSID and the duty cycle for the design activities.

The severe accident analysis will not be performed since with metal fuel, sodium coolant and pool heat capacity, core disruption accidents should belong in the "residual risk" category and therefore be relegated to probabilistic risk assessment domain. However, in preparation for Chapter 15 of the PSID, a sodium fire analysis will be performed to establish design measures against sodium combustion in the containment atmosphere and resulting thermal, deflagration or detonation loads that could challenge the integrity of the containment. These phenomenological analyses will be performed to support this task based on experimental data.

As details and refinements of the baseline design are developed to support the specific design, additional more detailed safety analyses will be pursued in the following years to provide input to the PSAR. A list and description of transient simulations beyond those considered in first year (including normal power maneuvering, anticipated operation sequences, design-basis events, as well as previously not considered design extension conditions) will be prepared.

Subtask 2.3 Probabilistic Safety Assessment (1mm)

During the first year, only the survey of the existing PSA methodologies will be carried with the goal of finalizing the PSA methodology in the early phase of the project

Subtask 2.4 Preliminary Safety Information Document

No activities are planned for this subtask during the first year.

Subtask 2.5 Preliminary Safety Analysis Report

No activities are planned for this subtask during the first year.

Task 3 Licensing Support (6mm)

The first year's effort for this task will be preparatory in nature. The design and analysis methodologies used at KAERI and ANL will be evaluated for the purpose of selecting the methodologies and the computer codes that will be utilized for the project. Neutronics codes are expected to be very similar between KAERI and ANL and hence it would be straightforward to select the reference computer codes to use. There is only a developmental computer code for the fuels performance modeling and hence a joint development might be a desirable outcome. Therefore, the major efforts will be in the area of the safety methodologies.

KAERI wishes to use ANL core neutronics codes for Korean SFR Prototype Design. The codes belonging to this category are MC2-3, DIF-3D with embedded VARIANT module, REBUS-3, VARI-3D with GPT capability and DIF-3D/K. For the harmonization with ANL code suites, these codes need to become available in KAERI as soon as possible.

Since ANL code suites will be subject to the KINS licensing review for the specific design approval, the available databases need to be decided. For this work, all the existing database need to be thoroughly reviewed in the view point of measured parameters, availability of measurement uncertainty, QA possibility, and identifying further experiment necessity. The plan for this effort will be initiated in the first year, but most work will need to be carried out in later years while interacting with KINS.

ANL developed SAS4A/SASSYS-1 has been the *de facto* standard worldwide and the first year's effort will be aimed at copyrighting the code by ANL and arranging a separate licensing agreement for the code.

표 5 ANL 업무기술서(Statement of Work) (FY13)

[redacted]

APPENDIX A

P-12021

KAERI-ANL Joint Program on Design Development of a Prototype Sodium-cooled Fast Reactor

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Statement of Work

KAERI-ANL Joint Program on Design Development of a Prototype Sodium-cooled Fast Reactor

1. Purpose

KAERI has been developing the KALIMER sodium-cooled fast reactor concept and recently decided to develop a 100 MWe size prototype in order to reduce the construction cost at the same time maintaining the design features desirable for commercialization. The most important innovation in the KAERI SFR is the utilization of metal fuel which enables unique inherent safety characteristics as well as a most efficient actinide burner.

The underlying key technology for the KAERI SFR is based on the fast reactor technologies developed at Argonne National Laboratory (ANL) in the 1980s and 90s. Therefore, a joint program between KAERI and ANL to develop a prototype SFR will be mutually beneficial. The joint program will be performed under a Work-for-Others Contract for the ANL portion of the work.

The purpose of this Statement of Work is to describe the ANL tasks in support of the joint program to develop a Specific Design in order to secure a Design Approval by the Korean licensing authority and the U.S. Nuclear Regulatory Commission for a future sodium-cooled fast reactor.

KAERI will be responsible for the overall SFR project and will have a much bigger role in all aspects of the design development and the licensing support activities than outlined here for the ANL portion. The ANL contributions will be fully integrated into the KAERI activities.

2. Task Description

Task 1 Specific Design of KAERI SFR

This task includes all design related activities associated with reactor core, reactor enclosure system, primary heat transport system, intermediate heat transport system, shutdown heat removal system, fuel handling system, instrumentation and control system, in-service inspection system, power conversion system, and buildings and structures as required to develop the Specific Design.

The design activities under this task will integrate key design features from KALIMER and past Argonne SFR designs and incorporate the lessons learned from the SFR operations worldwide.

Subtask 1.1 System Functional Requirements and Overall Process Flow Diagram

The system functional requirements will be developed in order to guide the structures, systems and components design activities. In addition, an overall process flow diagram will be developed for the reactor system, primary and intermediate heat transport systems, and main steam and feedwater systems. These system functional requirements and overall process flow diagram will be updated on an annual basis to take account of the design progression.

Subtask 1.2 Reactor Core Design

An optimized reactor core design specification that best meets the design goals will be developed through detailed design tradeoff studies involving the pin design parameters, burnup capability, fluence limits, thermal hydraulics, excess reactivity control, fueling interval, and other considerations. This subtask requires a close coordination with KAERI design activities in the early stages in order to establish a reference core design specification, which will form the basis for other design activities to proceed and also interface with the safety analysis activities. The activities under this subtask include detailed analyses and designs for the following:

- Tradeoff studies on power level and core design alternatives
- Reference core design determination
    - Reactivity control system
    - Steady-state thermal-hydraulics analysis
    - Kinetics parameters and reactivity coefficients as required for the safety analyses
    - Shielding
    - Fuel system design

Subtask 1.3 Reactor Enclosure System

Based upon the reference core envelope developed in Subtask 1.2, the reactor enclosure system design parameters and configurations will be developed. The design and engineering analyses will include the following major structures, systems, and components:

- Reactor vessel and support structure
- Guard vessel and support structure
- Reactor vessel head

All engineering analyses will be performed at nominal steady state conditions for anticipated normal loadings.

Subtask 1.4 Primary Heat Transport System

Based upon the reference core envelope developed in Subtask 1.2, the primary heat transport system design parameters and configurations will be developed. The design and engineering analyses will include the following major structures, systems, and components:

- Primary pump – conceptual specifications
- Intermediate heat exchanger – conceptual specifications
- Internal piping – sizing and design layout
- Lower internal structure – sizing and design layout
- Upper internal structure – conceptual specifications
- Core barrel
- Core restraint structure
- Redan

All engineering analyses will be performed at nominal steady state conditions for anticipated normal loadings. For the primary pump and intermediate heat exchanger, the work will include conceptual specifications including a thermal-hydraulic analysis to verify component sizing. Multi-dimensional thermal hydraulic analyses of reactor pool will be performed in conjunction with Task 2.

Subtask 1.5 Intermediate Heat Transport System

Consistent with the system functional requirements that are developed under Subtask 1.1, the intermediate heat transport system (IHTS) design parameters and configurations will be developed. This subtask will include development of:

- Overall piping specifications, sizing, and design layout, including expansion tanks for each loop,
- Steam generator conceptual specifications including thermal analysis to verify component sizing for the overall plant thermal load,
- Design and specifications for the Sodium-Water Reaction Pressure Relief System (SWRPRS) that safely depressurizes and drains the IHTS during a postulated steam generator tube rupture event,
- Sizing and specifications for the IHTS sodium pumps; both mechanical and electromagnetic (EM) pump designs will be considered as part of the task,
- Sodium purification (cold trap) design and specifications including plugging meters for quantitative evaluation of sodium purity level, and
- IHTS instrumentation layout and specifications.

Once the base design is developed, various analyses will be carried out to verify proper functionality under nominal steady-state conditions. Structural and thermal stress analyses will be carried out with ANSYS to verify acceptable stress levels in the system for the design lifetime of the plant. In addition, the structural integrity of the IHTS under a postulated large steam generator tube rupture event will be verified with the SWAAM sodium-water reaction code. This analysis will also provide the peak design pressure for the intermediate heat exchanger (IHX) design that will be developed under Subtask 1.4.

Subtask 1.6 Shutdown Heat Removal System

Consistent with the system functional requirements developed in Subtask 1.1, the shutdown heat removal system will be designed and analyzed. The preliminary specific design will include a trade study of the various options for decay heat removal and their impact upon reactor performance and reliability. Based upon this initial trade study, a SHRS approach will be decided and a preliminary specific design with appropriate engineering analyses performed that demonstrates the feasibility of the design under natural circulation conditions (loss of electrical power) and nominal steady state conditions. A description of the SHRS functions and requirements will also be prepared.

- Development of design concepts of shutdown heat removal system
- Evaluation of system and component performances

Subtask 1.7 Fuel Handling System

Based upon the reference core assembly size, the in-vessel and ex-vessel fuel handling mechanisms will be designed. The fuel handling system will include the following:

- In-vessel fuel handling mechanism and control system
- Rotatable plugs
- Ex-vessel fuel handling mechanism – physical size estimates, functions and requirements, and conceptual specifications
- Transfer cask technology – physical size estimates, functions and requirements, and conceptual specifications

Once the base design is developed, various engineering analyses will be performed at nominal steady-state conditions to verify proper functionality. Shielding requirements will also be determined for those systems located outside of the reactor vessel.

Subtask 1.8 Instrumentation and Control System

This work will focus on evaluating the various possibilities for making optimum use of the passive features and the inherent safety response in the design of the I&C system. The objective is to minimize safety channels and instrumentation in the plant protection system (PPS) and to utilize passive features and inherent feedbacks in the plant control system (PCS). The design activities will include:

- Steam generator leak detection and non-destructive examination techniques,
- Failed fuel detection and location system,
- Non-invasive instrumentation to minimize the number of instrumentation penetrations, thimbles and cabling,
- Component condition monitoring and diagnostics capabilities,
- Inherent response-based algorithms for plant controllers,
- PPS/PCS simulation to develop controller algorithms, instrument selection and location, and trip points.

- Instrumentation layout and specifications for the primary heat transport system, intermediate heat transport system, and shutdown heat removal system, if resources become available.

Subtask 1.9 In-Service Inspection

Innovations in inspection and maintenance technologies, e.g. under-sodium viewing, mechanical indexing and remote handling, will be evaluated for in-service inspection application. Innovative concepts may include ultrasonic sensor scanning for under-sodium viewing, safety measures in fuel handling, and partially automated remote handling and inspection devices. In order to enable the assessment of plant inspectibility and maintainability, a simulation environment will be developed, which allows test operation of in-service inspection and maintenance technologies in virtual environment. Integrating recent innovations in multi-modal display, distributed computer operating systems, and remote systems technologies and nuclear simulation technologies will allow accurate simulation to be conveyed in realistic multi-modal experience.

Subtask 1.10 Power Conversion System

The overall process flow diagram developed under Subtask 1.1 will include the power conversion system. Additional activities, such as plant operation strategy and control logics and plant performance analysis will be performed as the need arises.

Subtask 1.11 Buildings and Structures

The performance of the reactor building with and without a seismic isolation system will be analyzed. If it is decided to incorporate a seismic isolation system, then appropriate seismic isolation technologies for the reactor building will be recommended and incorporated into the reactor building design.

- Plant layout based on the reactor and fuel handling, and turbine building design
- Application of seismic isolation system for the reactor building and other structures as necessary

Task 2 Safety Analyses

This task consists of three main activities: development of the safety design criteria, safety analyses to be conducted in coordination with the design activities under Task 1, and preparation of safety documents to be submitted to the licensing authority.

Subtask 2.1 Establishment of Safety Design Criteria

The USNRC's General Design Criteria for Nuclear Power Plants (10 CFR Part 50 Appendix A) dictate the safety design and licensing of LWRs. There are no equivalent criteria for SFRs. Therefore, it is essential for the prototype SFR Project to take the initiative to develop and propose the safety design criteria, which will guide the safety design approaches as well as the licensing processes. The proposed safety criteria should be robust so that they form the foundation for the design activities, as well as being accepted by the licensing authority.

The ANSI Standards on SFR Safety Design Criteria and the NRC licensing reviews for the FFTF, CRBR, and PRISM Projects will be evaluated to recommend the Safety Design Criteria applicable for this project. The on-going activities of the ANS Standards Committee and the GIF Task Force will also be factored in. In particular, potential application of the risk-informed and/or performance-based regulatory approach will be evaluated in developing the Safety Design Criteria.

Consistent with the Safety Design Criteria, the next tier design-basis-events and beyond-design-basis-events will be developed, which need to be analyzed in detail in Subtask 2.2. DBEs and BDBEs adopted in the FFTF, CRBR, and PRISM Projects will form the base, however, a new set of DBEs and BDBEs will be developed considering the evolutions in design and safety approach.

Subtask 2.2 Establishment of PIRT

Phenomena Identification and Ranking Tables (PIRT) provide the basis for model development and experimental needs for safety analysis. ANL and KAERI will develop PIRT together by assembling the expertise from both organizations. It is recommended that more than 6 experts participate together in this task from various fields. It is also planned to identify the list of experiments or tests required in the future based on the established PIRT. A framework for the PIRT exercise will first be established. This will include PIRT topics, safety criteria, preliminary set of accidents, accident phases and scenarios, and phenomena. The framework will finalize the number of PIRT panel members/areas of expertise. Panel members will be identified and a Panel Chairman/facilitator will be appointed. Procedures for the conduct of the PIRT exercise will be established within this framework.

Subtask 2.3 Safety Analyses

The design-basis-events and beyond-design-basis-events developed in Subtask 2.1 will be analyzed using the Argonne developed SAS4A and SASSYS codes and other codes as necessary.

The bounding events will be identified that will establish the source term and the containment design basis.

In order to assure that the desired inherent safety features are built in to (not added on to) the design, Task 2 will be conducted in close coordination and consort with Task 1 so that the safety criteria are met though design modifications. Based on the trade studies which were completed years 1-2, the design options will have been selected and a baseline design will have been established. As details and refinements of this baseline design are developed to support the specific design, additional more detailed safety analyses will be pursued in years 2-4 to provide input to the PSID.

The analyses will cover the entire range of operating transients including normal power maneuvering, anticipated operation sequences, design-basis events, as well as beyond-design-basis events. As needed, the simulations will also cover the whole-plant dynamics including the detailed steam-generator model and energy conversion system response. The list of the transients (duty cycles) will include sufficient details to facilitate the modeling of the events with a dynamics code as well as an anticipated frequency of occurrence for each event. The list for transient analysis will be prioritized based on the PSA results from Subtask 2.3.

The results of the normal operational transients and design basis events will include consideration of uncertainties in the analysis to investigate the dynamics characteristics and performance of the entire plant, including the reactor and BOP sides. Uncertainties in the safety case evaluations and loading histories for the component design will be incorporated in the analyses either via utilization of hot channel factors or by employing an uncertainty quantification scheme based on sampling of input variables. Based on the results of the transients, the feedback on the reactor, BOP and control system design and functions will be provided. In addition, the results of the transients will be provided for duty cycle structural analyses of various components, such as the reactor vessel.

- Transient analysis for anticipated operation sequences (reactivity insertion, loss of core cooling, loss of normal heat sink)
- Analysis for design basis events (LOF, TOP, LOHS, partial flow blockage, pipe break, main vessel leak, sodium leakage, SG tube rupture, etc.)
- Analysis for beyond-design basis events (UTOP, ULOF, ULOHS, flow blockage, large SG tube rupture, sodium leakage)

Subtask 2.4 Probabilistic Safety Assessment (PSA)

Another important activity under the safety analyses is associated with the probabilistic safety assessment (PSA). An appropriate methodology for PSA will be developed in the early phase and Level-1 and Level-2 quantification will be followed. Using preliminary design features, an initial model can be developed and analyzed to identify which accident sequences could be candidates for more detailed study and analysis. The model can be fine-tuned and expanded to include more event types (such as external events) with more site-specific details.

- Finalization of PSA methodology
- Establishment of the basis for mechanistic source term
- Level-1 PSA
- Level-2 PSA

Subtask 2.5 Preliminary Safety Information Document (PSID)

As sufficient results have been obtained from the safety analyses activities under subtask 2.2 and design iterations have been completed with the Task 1 activities, the preliminary safety information document (PSID) will be prepared. The purpose of the PSID is to submit it to the Korean licensing authority as the basis for a preliminary determination of licensability and assessment of safety issues in the design.

Subtask 2.6 Preliminary Safety Analysis Report (PSAR)

The preliminary safety analysis report (PSAR) will be developed incorporating the findings in the Safety Evaluation Report for the PSID.

Task 3 Licensing Support

Once the PSID is submitted to the licensing authority, there certainly will be technical questions raised, which would require clarifications or additional analyses. This task will provide the technical support during the licensing review process. More importantly, the computer codes and the database used in the safety analyses will be verified and validated as described in the following subtasks.

Subtask 3.1 Validation of Neutronics Design Computer Codes

The neutronics methodologies, including multi-group cross section generation, whole-core neutronics calculations and depletion calculations, will be validated against appropriate benchmark problems and/or appropriate critical experiments data.

It is assumed that the physics database used for the validation of ANL neutronics codes will be available to Korea Institute of Nuclear Safety upon request during the licensing review of corresponding topical reports.

Subtask 3.2 Validation of Fuel Design Basis

This subtask will document the U.S. experience with metallic fuel (U-Zr and U-TRU-Zr) and fuel components (cladding and duct), in particular its irradiation performance in EBR-II, FFTF, and TREAT, including steady state, transient and run-beyond cladding breach performance.

Existing models for important phenomena related to fuel performance such as FCCI, constituent redistribution, fission gas induced swelling, thermo-mechanical models, etc., will be evaluated and in collaboration with KAERI an advanced fuel performance code will be developed and validated against the EBR-II and FFTF experimental database.

In addition, existing out-pile experimental data will be evaluated and will assist KAERI in planning execution of additional out-of-pile experiments.

Subtask 3.3 Validation of Safety Analyses Methodology

The SAS4A/SASSYS-1 code system has been used extensively as a design basis analysis tool for the EBR-II, FFTF and CRBR reactors, as a conceptual design evaluation code in the U.S. DOE reactor development projects (e.g. LSPB, SAFR, PRISM) as well as in severe accident calculations for the FFTF and CRBRP reactors in the U.S., for the SNR-300 reactor in Germany, for the MONJU reactor in Japan, for the PEC reactor in Italy, and for the BN-600 reactor in Russia. The models in SAS4A/SASSYS-1 have been validated with extensive analyses of laboratory experiments and in-pile fuel tests in the TREAT and CABRI as well as with reactor and plant test data from EBR-II and FFTF. An updated compilation of the past verification and validation reports of SAS4A/SASSYS-1 will be prepared.

These international validation bases will be extended to support the design and licensing of the KAERI SFR. Since the SAS4A/SASSYS-1 code system has been developed primarily as a design analysis tool, using it as a licensing tool would require extensive effort for compliance of software quality assurance practices required by national licensing authorities. Therefore, bulk of the task will be devoted to collaboration with KAERI staff to place the SAS4A/SASSYS-1 code system under configuration control, to establish a vigorous code verification and QA plan for code maintenance, and testing of software components through improved software quality engineering practices.

The creation of an input and post processor as an improved user interface to reduce potential input errors can also be performed under this task in collaboration with KAERI analysts. Further improvements to support parallel applications by taking advantage of standard parallel computing platforms can allow utilization of the SAS4A/SASSYS-1 code system as the simulation engine for the automated design optimizations, as well as the uncertainty quantification and sensitivity analysis schemes. If the design of the KAERI SFR is to withstand the regulatory scrutiny especially in at the specific design stage, the software system that supports the license application will likely be required to have these capabilities in place.

3. Computer Codes to be used for the ANL Tasks

There are 4 categories of computer codes that Argonne personnel plan to use in the execution of the ANL tasks:

1. ANL codes available (or planned to be) from RSICC or NEA Data Bank
   - ETOE/MCC-3 (update of MCC-2)
   - DIF3D
   - REBUS-3
   - SWAMM
2. ANL codes not available from RSICC or NEA Data Bank – DOE's "Applied Technology"
   - SAS4A/SASSYS-1
3. ANL codes that will be further developed during this project

- VARI3D
- LIFE-METAL
- NUBOW-3D

4. Commercial codes with ANL interface routines
   - SUPERENERGY-2-ANL
   - STAR-CD/STAR-CCM+
   - ANSYS
   - SAP2000

KAERI already has access to the computer codes in Category 1 above.

If KAERI desires to have access to any computer code in Category 2, an explicit written approval by DOE-NE is required since they are categorized as DOE's "Applied Technology." ANL will be responsible to secure the DOE approval.

The computer codes in Category 3 have not been fully developed for release. However these codes will play an important role for the ANL tasks, and hence the development will be resumed for ultimate use for the project. When these codes are ready for release and if KAERI desires to have access, the same procedure discussed above for Category 2 will apply here as well. KAERI is welcome to participate in these code development activities, at KAERI's expense, however the final coding changes and documentation will be the responsibility of ANL in order to maintain the necessary quality assurance.

If KAERI wishes to use any of the Category 4 computer codes, it is KAERI's responsibility to acquire the license for these codes. ANL will share the interface routines and experiences.

4. Project Schedule and Milestones

The project schedule and major milestones are as follows:

- Project Duration: March 1, 2012 through February 28, 2021
- Complete Conceptual Design: February 28, 2013
- Complete Preliminary Safety Information Document: February 28, 2016
- Complete Preliminary Design: February 28, 2016
- Complete Safety Analysis Report: February 28, 2018
- Complete Specific Design: February 28, 2018
- Complete Design Approval by Licensing Authority: February 28, 2021

5. Project Efforts and Costs

The Project efforts (in terms of man-months) are summarized in the table below in subtask levels for the first 5 years.

	Year 1	Year 2	Year 3	Year 4	Year 5
1. Specific Design					
1.1 Requirements	2	3	2	2	2
1.2 Reactor Core	8	13	14	6	6
1.3 Enclosure	9	7	16	14	13
1.4 PHTS	10	3	24	22	22
1.5 IHTS	6	2	21	20	19
1.6 SHRS	6	9	15	14	12
1.7 Fuel Handling	9	5	21	20	18
1.8 I&C	2	14	6	6	6
1.9 ISI	1	0	4	4	4
1.10 PCS					
1.11 Buildings	2	6	6	6	6
2. Safety Analyses					
2.1 Design Criteria	2	3	2	2	2
2.2 PIRT	0	6			
2.3 Safety Analyses	7	6	22	24	24
2.4 PSA	1	5	13	17	20
2.5 PSID			14	24	
2.6 PSAR					24
3. Licensing Support					
3.1 Neutronics	1	1	13	13	13
3.2 Fuel Basis	2	4	13	13	13
3.3 Safety	0	1	13	13	13
Total (man-month)	68	88	219	220	217

The total project costs (in thousand dollars) are summarized in the table below.

	Year 1	**Year 2**	**Year 3**	**Year 4**	**Year 5**
Salaries	992	1,228	3,215	3,306	3,328
Fringe Benefits	199	246	821	864	854
Indirect Costs	409	506	1,564	1,630	1,618
M&S, Travel	100	100	200	200	200
Total	**$1,700**	**$2,080**	**$5,800**	**$6,000**	**$6,000**
	Year 6	**Year 7**	**Year 8**	**Year 9**	**Total**
Total Salaries	3,289	3,226	3,169	3,169	24,922
Fringe Benefits	864	903	936	936	6,623
Indirect Costs	1,647	1,671	1,695	1,695	12,438
M&S, Travel	200	200	200	200	1,800
Total	**$6,000**	**$6,000**	**$6,000**	**$6,000**	**$45,780**

The annual budget presented above is subject to change based on the detailed work plan to be developed for each year and the funding availability. ANL will develop, in consultation with KAERI, the detailed work plan for each year by January 1. The annual budget will be adjusted as necessary by mutual agreement and the WFO Agreement will be amended as necessary.

6. Project Meetings

The joint project meetings will be scheduled on a regular basis (between 2 to 3 months) alternating the meeting places between KAERI and ANL to review the technical progress and to decide on the future action items.

7. KAERI Assignees at ANL

For the purpose of joint participation in the project and to fulfill the liaison role, a mutually agreed upon number of KAERI staff can be assigned to stay at ANL for the duration of up to one year for each assignee. KAERI will be responsible for the entire cost of assignment including salaries, travel and living costs, and health and other insurance costs. ANL will provide the office facilities with desktop computers and supplies.

8. Project Deliverables

The deliverables will include:

- ANL contributions to Functional Requirements and Specifications Document
- ANL contributions to Conceptual Design Report
- ANL contributions to Preliminary Safety Information Document
- ANL contributions to Preliminary Design Report
- ANL contributions to Safety Analysis Report
- ANL contributions to Specific Design Report
- Computer codes as agreed to in Section 3.

In addition, other deliverables can be defined at the regularly scheduled project meetings based on mutual agreements.

Supplement to
APPENDIX A
Statement of Work

Consistent with the Statement of Work, this Supplement is prepared to elaborate on more details of the Argonne activities to be performed under the WFO Agreement.

1. Relationship between the work performed at ANL under this Statement of Work and the activities carried out at KAERI

The Statement of Work attached to the Work for Others (WFO) Agreement describes the activities performed by Argonne National Laboratory in support of the KAERI's Prototype Sodium-cooled Fast Reactor (SFR) Project. Therefore, the KAERI's activities are not included in this document. However, the KAERI's SFR Project will be a joint effort between KAERI and ANL for the workscope described in the Statement of Work. The design will be developed jointly; the safety analyses and documents submitted to the licensing authority will be jointly developed; and the defense in licensing will also be a joint effort. However, KAERI will be responsible for the overall SFR project and will have a much bigger role in all aspects of the design development and the licensing support activities than outlined in Appendix A for the ANL portion. The ANL contributions will be fully integrated into the KAERI activities. Further, KAERI shall have unlimited rights (right to use, duplicate or disclose, and permit others to do so) to ALL technical data produced by ANL in the performance of the work under this agreement.

The most important factor for the ANL's participation in this joint effort is due to the ANL's unique qualifications. Argonne's qualifications are unique in two respects. First, ANL designed, constructed, and operated Experimental Breeder Reactor-I (EBR-I) and EBR-II. Furthermore, ANL was extensively involved in the Clinch River Breeder Reactor (CRBR), Fast Flux Test Facility (FFTF), Large Scale Prototype Breeder (LSPB), Large Pool Plant (LPP), Sodium Advanced Fast Reactor (SAFR), and Power Reactor Innovative Small Module (PRISM) projects both in design and licensing support activities. Recently ANL also carried out the Small Modular Fast Reactor (SMFR) in collaboration with Japan Atomic Energy Agency and French Atomic Energy Commission and Advanced Burner Test Reactor (ABTR) in support of the Global Nuclear Energy Partnership (GNEP) initiative. ANL will incorporate the lessons learned from these numerous SFR projects into the KAERI SFR Project to make it a model prototype. The SFR experience worldwide has a mixed record of operation. Understanding what has worked, what has not worked, and why and applying them from the onset of the conceptual design is crucial. Argonne designers bring such expertise.

In addition, Argonne's experience in licensing of the past SFRs led to its conviction that for the future SFRs to be viable, inherent passive safety has to be exploited with emphasis on prevention of severe accidents than the mitigation features and myriad of engineered safety systems. Argonne will support KAERI to bring about a new paradigm in licensing approach for this SFR Project.

The other aspect of Argonne's unique qualifications is associated with its R&D accomplishments during the Integral Fast Reactor (IFR) program in the 1980s and 90s. The R&D investments during these periods exceed $1 billion, including the operating costs of the supporting facilities. The metal fuel development was indeed the foundation of the IFR program. The extensive database developed by Argonne on the metal fuel irradiation performance and its safety characteristics would be essential for the preparation of the safety documents to be submitted to the licensing authority and for a successful design approval process. The creation of the equivalent fuels and safety database by KAERI alone would require more than a decade of concentrated R&D efforts and extensive resources.

2. Detailed Plan for ANL Activities in FY2012 and FY 2013

The main objective of FY2012 activities is to complete a conceptual design. Because of a delayed start of the ANL activities, a series of tradeoff studies will be performed to evaluate the advantages and disadvantages of the design features in the KAERI-developed conceptual design in comparison to alternative design features. This will cause some design changes in the early part of the preliminary specific design.

The main objective of FY2013 activities is to start a preliminary specific design, which will form a reference design and starting point for improving the conceptual design: (1) further optimization and design tradeoffs of each system and subsystems, (2) implementation of integrated safety analyses, and (3) evaluation of the system thermal-hydraulic and transient analyses.

Task 1 Specific Design of KAERI SFR (19mm in FY12 and 98mm in FY13)

Subtask 1.1 System Functional Requirements and Overall Process Flow Diagram (1mm in FY12 and 4mm in FY13)

KAERI will update the fast reactor system functional requirements developed in a conceptual design phase, which will guide the structures, systems and components preliminary specific design activities, and ANL will review the system functional requirements and provide comments. This task will also include the development of an overall process flow diagram for the reactor system, the primary and intermediate heat transport systems, and main steam and feedwater systems. KAERI will develop overall process flow diagram. ANL will review the overall flow diagram of KAERI. The functional requirements will be updated on an annual basis to take into account for the design progression.

Deliverables:
Review comments for system functional requirements for conceptual design phase (1.31)
Review comments for overall process flow diagram (4.30)
Review comments for system functional requirements for updated functional requirements (6.30)

Subtask 1.2 Reactor Core Design (5mm in FY12 and 16mm in FY13)

Establishing a well optimized core design is crucial for the success of the overall reactor design because the design of the rest of the reactor enclosure system, fuel handling system and so on depends on the core envelope and the safety analyses cannot be initiated without the reference core specifications.

A tradeoff study on the Core Design Options will be performed to compare the core design developed by KAERI during the conceptual design phase with alternative designs. The evaluation will include the effects of core outlet temperature, cladding material, linear heat rating, pressure drop, as well as other design constraints and parameters on the overall core performance characteristics.

Based on the tradeoff study, KAERI and ANL will jointly define an optimized reference core design and more detailed analyses will be performed to fine tune the reference design including:

- Cycle-by-cycle fuel management strategy
- Reactivity control analysis
- Reactivity coefficients as required for safety analysis
- Evaluation of uncertainty factors
- Shielding analysis
- In-vessel storage location

Deliverables:
Preliminary result of tradeoff study (1.31)
Final report of tradeoff study (2.28)
Report on core performance characteristics and fuel management strategies (12.31)
Report on shielding analysis (12.31)

Core Restraint System: This is a new subtask transferred from Subtask 1.3. The core restraint system study will include information on common choices for a core restraint system and discuss the pros and cons of each system. This study will evaluate the systems' performance with regard to the functions of:

- Providing core alignment
- Limiting step reactivity insertion
- Maintaining negative reactivity feedback during power ascent
- Accommodating irradiation induced creep and swelling effects
- Maintaining refueling loads below specified limits.

The design of the core restraint system affects the bowing deformation of a core and has a significant effect on the inherent safety of the system. The performance of the system depends on a number of design variables in the core which dictate the amount of bowing and the structural loads that result from duct contact. This study will include a discussion regarding these parameters and how they affect the design.

Another task is to develop the preliminary design of the core restraint system for the reference design. The core restraint performance depends on the structural response of the core assemblies and depends upon many variables including the duct geometry, duct material, number and location of load pads, stiffness of the nozzle support. The activities associated with the core restraint design include:

- Design and analysis of fuel assembly ducts (material, geometry, load pad dimensions and position, and nozzle hold-down mechanism)
- Design and analysis of the core restraint rings (material, geometry and mounting details)
- Design and analysis of the nozzle support (nozzle stiffness/compliance and bearing points, hydraulic hold-down mechanism)

The design will be achieved through iterative process. The analysis of the overall core restraint system will be done with the NUBOW-3D code. The analysis has two main goals: (1) to ensure negative reactivity feedback due to power/flow ratio increase and (2) to ensure that refueling loads are within acceptable limits. The reactivity change is driven by thermal deflections of the core assemblies. To ensure negative reactivity feedback, the analysis is run at various times in the fuel cycle to assess the response of the core to a power/flow ratio increase. The reactivity change due to the bowing configurations for each value of P/F ratio is determined. The refueling loads are driven by the permanent inelastic deformations that remain in the assemblies at the refueling temperature. These deformations are the result of irradiation creep and swelling. To assess the refueling loads, the analysis is run over the life of the core and the permanent deformations and resulting contact loads are assessed at the refueling point to confirm that they are in acceptable limits. The various design parameters which control the core bowing response are iterated until a satisfactory response is obtained.

The core restraint analysis requires as input from the core physics, the system description of the core, temperature at normal operating power, flux profile, and reactivity displacement worth. The design of the core restraint system is coupled with the core support structures, the fuel handling machine, and the core design. The core restraint rings are contained within and supported by the core barrel. The fuel assemblies are supported by the core assembly receptacles attached to the upper and lower grid plates. The fuel sub-assembly duct material and inner hexagonal dimensions depend upon the core physics design. The fuel handling system dictates the allowable refueling loads. The analysis and design of the core restraint systems will be done iteratively and in close communication with these interfacing systems.

Using the loading information obtained from NUBOW, more detailed analysis and design of the assembly ducts and restraint system components. Stress analysis will be done with respect to the AMSE BPV Section III wherever applicable. Additional analysis will be done using the ANSYS software.

Deliverables:
Report on Approach to Core Restraint systems (2.28)
Report on the preliminary core restraint system design, including analysis results, system design specifications, and drawings illustrating key assembly and component dimensions (14.2.28)

Subtask 1.3 Reactor Enclosure System (4mm in FY12 and 12mm in FY13)

In lieu of the original FY12 scope of work, a major tradeoff studies will be performed as described below.

Tradeoff Study on Reactor Vessel Internal Redan and Heat Transport Path configuration: This study will examine a reconfiguration of the prototype SFR concept from a hot pool DRACS concept to a cold pool DRACS concept and the associated reduction in physical size of the reactor vessel. This work will be performed in two steps, the first step is to provide a re-orientation of the prototype SFR heat transport system to a cold pool DRACS. The section step will be a design analysis of this configuration to show that decay heat can be removed and passive safety can be maintained in this configuration.

In addition, the reactor vessel support structure will be designed including the support flange design and bolt design in compliance with the ASME BPV III, Division 5 rules.

In addition to the tradeoff studies discussed above, ANL will carry out the following item based upon the KAERI conceptual design of the reactor enclosure system developed in the first year:

- Reactor vessel support structure –This work will include the support flange design and bolt design in compliance with the ASME BPV III, Division 5 rules.

Any initial engineering analyses will be performed at nominal steady state conditions for anticipated loadings. ANL will review the reactor vessel support structure report and provide comments.

Deliverables:
Review of Reactor Enclosure System Conceptual Design Report (1.31)
Preliminary comparison of cold pool DRACS with re-orientation of the reactor internal structures (2.28)
Final design analysis of cold pool DRACS system with re-oriented reactor internal structures (6.30)
Reactor vessel support structure design report (10.31)

Subtask 1.4 Primary Heat Transport System (3mm in FY12 and 10mm in FY13)

Fluid System Development (1mm in FY12 and 2mm in FY13)

System design requirements and component design requirements will be developed by KAERI to preliminary specific design level. The related design works include;

- Preparation of document for system design requirements of PHTS by KAERI
- Preparation of document for component design requirements of PHTS by KAERI – including sizing of components such as PHTS pump and IHX

By ANL, the design requirements in above documents will be reviewed and checked, and the review reports will be prepared with comments and supplementation.

Deliverables:
Review and comments on conceptual design of IHX (1.31)
System design requirements of PHTS by KAERI (8.31)
Component design requirements of PHTS by KAERI (8.31)
Review of system design requirements of PHTS by ANL (10.31)
Review of component design requirements of PHTS by ANL (10.31)

Mechanical Structure Development (2mm in FY12 and 8mm in FY13)

KAERI will provide conceptual design information for their PHTS concept for ANL review and feedback.

Based upon the KAERI conceptual design of the primary heat transport system (PHTS) developed in the first year, ANL jointly with KAERI will carry out the following items:

- Primary pump – proper size of the driving shaft length and diameter with considerations of the rotor dynamic characteristics by the preliminary analyses.
- Receptacle within inlet plenum – sizing and design layout of the receptacle along with the space between the two grid plates.
- Upper internal structure – arrangement design of the thermocouple drywell guide tubes and the CR shroud tubes including the flow penetrations on the guide plates.
- Control Rod Drive Line – preliminary sizing of a control rod drive line including the shielding method of the CR drive line and the tension tube

Deliverables:
Review of the PHTS conceptual design (1.31)
PHTS structure design report to include inlet plenum receptacle, UIS TC arrangement, and CRDL sizing (10.31)

Subtask 1.5 Intermediate Heat Transport System (8mm in FY13)

System design requirements and component design requirements for IHTS and Sodium-Water Reaction Pressure Relief System (SWRPRS) will be developed by KAERI to preliminary pre-specific design level. The related design works include:

- Preparation of document for system design requirements of IHTS and SWRPRS by KAERI
- Preparation of document for component design requirements of IHTS and SWRPRS by KAERI – including sizing of components such as IHTS pump and steam generator

By ANL, the design requirements in above documents will be reviewed and checked, and the review reports will be prepared with comments and supplementation.

In addition, the sizing and specification for cold trap to purify the intermediate sodium will be developed to preliminary pre-specific design level by ANL.

Deliverables:
Review of Intermediate Heat Transport System Design (1.31)
System design requirements of IHTS and SWRPRS by KAERI (8.31)
Component design requirements of IHTS and SWRPRS by KAERI (8.31)
Report for sizing and specification of cold trap by ANL (8.31)
Review of system design requirements of IHTS and SWRPRS by ANL (10.31)
Review of component design requirements of IHTS and SWRPRS by ANL (10.31)

Subtask 1.6 Shutdown Heat Removal System (3mm in FY12 and 12mm in FY13)

ANL will provide review of SHRS design concept and evaluate potential alternatives between a hot pool DRACS system and a cold pool DRACS system. ANL review will include sizing calculations.

System design requirements and component design requirements will be developed by KAERI to preliminary pre-specific design level. The related design works include;

- Preparation of document for system design requirements of SHRS by KAERI
- Preparation of document for component design requirements of SHRS by KAERI – including sizing of components such as heat exchangers and expansion tank

By ANL, the design requirements in above documents will be reviewed and checked, and the review reports will be prepared with comments and supplementation.

In addition, the sizing and specification for EM pump in the active SHRS will be developed to preliminary pre-specific design level by joint work led by ANL.

Deliverables:
Review of SHRS conceptual design (1.31)
Evaluation of hot pool vs. cold pool DRACS system (5.31)
System design requirements of SHRS by KAERI (8.31)
Component design requirements of SHRS by KAERI (8.31)
Report for sizing and specification of EM pump in the active SHRS by ANL (8.31)
Review of system design requirements of SHRS by ANL (10.31)

Review of component design requirements of SHRS by ANL (10.31)

Subtask 1.7 Fuel Handling System (2mm in FY12 and 12mm in FY13)

Based upon the KAERI conceptual design of the fuel handling system developed in the first year, ANL jointly with KAERI will carry out the following items:

- Design requirements and design descriptions for in-vessel fuel transfer machine (IVTM)
- Design requirements and design descriptions for ex-vessel fuel transfer machine (EVTM)
- Conceptual design of adapter structures between EVTM and fuel transfer port
- Provide lists of detailed design documents of fuel handling system for ABTR and PRISM

KAERI will need to provide ANL with the necessary design data for the conceptual design for the fuel handling system for ANL to carry out this work.

In addition, a tradeoff study will be performed to compare the single rotatable plug/pantograph and double rotatable plug/straight-pull fuel handling systems.

Deliverables:
Review of KAERI conceptual design of FHS (1.31)
Lists of detailed fuel handling system design documents (4.31)
Tradeoff study on single vs. double rotatable plug fuel handling systems (6.30)
Fuel handling system design report (12.31)

Subtask 1.8 Instrumentation and Control System (16mm in FY13)

This work aims for establishing the I&C design requirements of SFR to preliminary pre-specific design level by joint work led by ANL. Also, the instrumentation methodology, specification and layout should be developed for important instrumentation systems. The design activities will include:

- Preparation of documents for preliminary I&C design requirements of SFR by ANL
- Preparation of documents for I&C design requirements of SFR by KAERI with ANL support
- Review and check for I&C design requirements of SFR by ANL
- Instrumentation methodology, specification and layout for failed fuel detection and location system by ANL
- Instrumentation methodology, specification and layout for steam generator leak detection system by ANL

Deliverables:
Preliminary I&C design requirements of SFR by ANL (5.31)

I&C design requirements of SFR by KAERI with ANL support (10.31)
Review report for I&C design requirements of SFR by ANL (11.30)
Report for instrumentation methodology, specification and layout of failed fuel detection and location system by ANL (14.2.28)
Report for instrumentation methodology, specification and layout of steam generator leak detection system by ANL (14.2.28)

Subtask 1.9 In-Service Inspection (1mm in FY12)

ANL will review the ISI document developed for the conceptual design report. No ANL activities are planned for FY13.

Subtask 1.10 Power Conversion System

No ANL activities are planned for FY13.

Subtask 1.11 Buildings and Structures (8mm in FY13)

Based upon the KAERI conceptual design of the reactor building arrangements developed in the first year, ANL will carry out the following items;

- General building layout and sizing with consideration of seismic isolations, fuel transfer, component supports, and component repair & replacements concept.
- Establishment of containment boundary concepts
- Interface requirements for the piping system between seismic isolation and non-isolation buildings

KAERI will need to provide ANL with the necessary design data for the conceptual design and major components for ANL to carry out this work.

Deliverable:
Provide Information on seismic isolation technology (1.31)
Reactor building arrangement design report (10.31)

Task 2 Safety Analyses (3mm in FY12 and 27mm in FY13)

The FY12 safety tasks will focus on the preparation of the safety design approach document which exploits the inherent safety potential for accident prevention than emphasis on mitigation features. In FY13 it is proposed to enhance the studies on safety design criteria, PIRT, PSA methodology and reliability database to prepare the basis for the code systems and model validation used in the safety analysis and safety case. It is also planned to perform some safety analyses to evaluate the safety characteristics of the conceptual design for various postulated conditions. The ultimate goal of the safety analysis will be to show that radioactivity release to the environment can be prevented even during the most unlikely double fault events, and to ensure that the safety is built-in (as opposed to added-on) to the design. This approach will be followed in subsequent years.

Subtask 2.1 Establishment of Safety Design Criteria (3mm in FY12 and 2mm in FY13)

Essential to the establishment of safety design criteria, is the prior selection of the safety design approach. The emphasis here is on the necessity for a paradigm shift in the licensing process to one based on the risk-informed, performance-based regulatory approach. In this regard, it should be pointed out that the (1) NRC Policy Statement on Advanced Reactors, which emphasized prevention over mitigation and the importance of passive safety, (2) NRC plan or intention for 10CFR Part 53 (Technology Neutral Framework), and the (3) ANSI 54.1 activity are all trending in the direction of this paradigm shift. Based on this new trend for advanced reactors, the Project should take a leadership role in establishing a new safety design approach going beyond the traditional defense-in-depth approach by exploiting inherent safety and severe accident prevention. In FY12, the efforts will be focused on documenting the whats, hows and whys of this paradigm shift in the licensing process to a risk-informed, performance-based regulatory approach can lead to benefits in the safety design approach if properly utilized. It will show how the experimental evidence to support this new approach is available in EBR-II tests, TREAT tests, EBR-I post-mortem as the safety approach moved toward passive safety and away from mitigation.

Further investigations and considerations on safety design criteria will be pursued in FY13. A safety design criteria document will be developed in coordination with KAERI's conceptual design and also by incorporating the safety analysis for the design concept. The main focus would be on the following issues:

- Passive design and its impact on safety
- Prevention and mitigation of severe accident consequences
- Reduction of sodium risks
- Safety limits

Included in the considerations on safety design criteria will be the source term and design bases such as the containment design basis.

Deliverables:
Report on safety design approach (2.28)
SFR safety design criteria and evaluation report (12.31)

Subtask 2.2 Establishment of PIRT (6mm in FY13)

Phenomena Identification and Ranking Tables (PIRT) provide the basis for model development and experimental needs for safety analysis. ANL and KAERI will develop PIRT together by assembling the expertise from both organizations. It is recommended that more than 6 experts participate together in this task from various fields. It is also planned to identify the list of experiments or tests required in the future based on the established PIRT. A framework for the PIRT exercise will first be established. This will include PIRT topics, safety criteria, preliminary set of accidents, accident phases and scenarios, and phenomena. The framework will finalize the number of PIRT panel members/areas of expertise. Panel members will be identified and a Panel Chairman/facilitator will be appointed. Procedures for the conduct of the PIRT exercise will be established within this framework. The first PIRT panel meeting is envisioned at the end of March or beginning of April.

Deliverables:
Complete framework for the PIRT exercise (2.28)
PIRT for a metal-fueled pool-type SFR (12.31)
Summary of existing experiments or test data and identification of required tests or experiments (12.31)

Subtask 2.3 Safety Analyses (13mm in FY13)

ANL will analyze the sodium-water reaction accident with the SWAAM-II code, developed by ANL, for the conceptual design developed by KAERI in the first year. The code methodologies required for the analysis and background information used for the code development have previously been shared between ANL and KAERI. KAERI will need to provide ANL with the necessary design data for well-defined plant systems all the way to the /steam generator. A detailed layout of the IHTS will be required with the details of the piping and the multiple junctions if any, the surge tank, sodium-water reaction relief systems, components such as the IHX and the steam generator tube bundle. Steady state thermal-hydraulic conditions for the IHTS and the steam generator bundle water-side will also be needed. Collaboration with KAERI will be required to define the specific accident scenario.

Additional collaboration is expected on the analyses of containment performance for sodium fire and DBE analyses. The severe accident analysis will not be performed since with metal fuel, sodium coolant and pool heat capacity, core disruption accidents should belong in the "residual risk" category and therefore be relegated to probabilistic risk assessment domain. However, in preparation for Chapter 15 of the PSID, a sodium fire analysis will be performed to establish design measures against sodium combustion in the containment atmosphere and resulting thermal, deflagration or detonation loads that could challenge the integrity of the containment. These phenomenological analyses will be performed to support this task based on experimental data. Collaboration will be required with KAERI to define the specific accident scenario for the conceptual design developed by KAERI in the first year.

The quantification of uncertainty for reactivity models and other models will also be investigated. Some parametric analyses will be performed for selected categories of accidents. The most challenging accidents are so-called unprotected events where the second fault includes the failure of the reactor protection system to shut down the fission reaction in the core. The first fault can be any event that introduces significant misbalance between the heat produced by the reactor and the ability of the cooling system to remove this heat from the core. The past experience has shown that the most severe results are expected for the unprotected event of the primary coolant pump failure (so-called unprotected loss-of-flow, or ULOF, event), unprotected loss-of-heat-sink (ULOHS) event, or unprotected control rod ejection (unprotected transient over power, or UTOP, event). Those anticipated transients without scram have been historically the focus of the SFR safety analysis, and they will also be given the priority for the safety analysis during FY13. These are the categories of accidents for which the quantification of uncertainty for reactivity models and other models will be investigated. The parametric analyses will be performed for these selected categories of accidents.

Deliverables:
Sodium-water reaction analysis report (14.2.28)
Investigation of sodium fire and its analysis (12.31)
Parametric safety analyses report (12.31)

Subtask 2.4 Probabilistic Safety Assessment (6 mm in FY13)

In FY13, it is planned to investigate a PSA methodology for event frequencies applicable to metal-fueled SFR design analysis and safety analysis. The strategy for the preparation of a reliability data base is another important topic to provide meaningful PSA results. In FY13, ANL will carry out the survey of the existing methodologies with the goal of finalizing the methodology in the early phase of the project together with KAERI.

Deliverables:
Survey of PSA methodology and strategy for the development of reliability database (12.31)

Subtask 2.5 Preliminary Safety Information Document

No activities are planned for this subtask in FY12 and FY13.

Subtask 2.6 Preliminary Safety Analysis Report

No activities are planned for this subtask in FY12 and FY13.

Task 3 Licensing Support (1mm in FY12 and 8mm in FY13)

As part of the licensing process, the tools utilized in the preliminary design process will need to be validated – confirming both the computer code analysis techniques and the associated database. For key safety aspects, it will be important to clearly identify the validation status in a timely manner.

A variety of computer codes will be utilized by Argonne in various design phases. Furthermore, this Argonne contribution reflects both U.S. fast reactor database information and best design practices based on SFR experience. Because these tools represent a significant investment by US-DOE in fast reactor technology, they are categorized as "Applied Technology" and explicit approval by the Department of Energy will be required for release within this Project, with the benefits of U.S. tool utilization in the licensing process a primary motivation.

In this task, the licensing relevant computational tools and associated validation databases will be identified; and a prioritized schedule for licensing support will be developed by the following procedure:

1. Consultation with KAERI design team on what tools will be utilized for design
2. Prioritization of which tools and databases will be most important for the licensing support and safety confirmation
3. Creation of a schedule for assuring the proper tools and validation information is available for the licensing phase

Subsequent efforts (e.g., Subtask 3.2) will be required to prepare and finalize the validation packages for licensing of a prototype SFR. In addition, this schedule is a first step to securing timely approval for release of U.S. tool and database needs.
For each code, a status report will be created by the cognizant Argonne fast reactor analysis expert. This report will identify the current status of the computer code (e.g., copyrighted for external distribution), briefly describe the computational techniques, and identify the general content and status of associated U.S. databases. Finally, the effort required to prepare the codes and database materials for utilization in a licensing process will be clearly identified.

A similar procedure will have to be followed for the safety and fuels experimental database as well since these are also "Applied Technology."

Subtask 3.1 Validation of Neutronics Design Computer Codes (1mm in FY12 and 1mm in FY13)

A variety of neutronics analysis tools have been identified for design application, including the MC^2-3 code which will be requested through the copyright license procedure. The validation of these reactor physics tools relies heavily on critical experiments that were conducted concurrent with code development. To start the validation activity, a summary list of relevant critical experiments will be compiled and delivered to KAERI.

Deliverables:
Report on identification and status of critical experiments data (1.31)

Subtask 3.2 Validation of Fuel Design Basis (6 mm in FY13)

ANL will document the U.S. experience with U-Zr metallic fuel, in particular its irradiation performance in EBR-II including steady state, transient and run-beyond cladding breach performance. The task will provide for description of existing metallic fuels database.

ANL will establish preliminary fuel design criteria and use ANL fuel performance code (LIFE-METAL) to provide preliminary evaluation of the reference fuel design and suggest fuel design modifications as needed. The task will include description of the analysis methodology of LIFE-METAL code and evaluation of proposed design using preliminary design criteria.

Deliverables:
Summary of metal fuel data base (8.30)
Preliminary report on metal fuel performance analysis (11.30)

Subtask 3.3 Validation of Safety Analyses Methodology (1 mm in FY13)

ANL developed SAS4A/SASSYS-1 has been a unique tool to demonstrate the early termination of initiating phase of severe accident in metal-fueled SFRs. It is necessary to share the common understanding of the capability of the code and model validation for future licensing based on the experimental data and analysis results.

This SAS4A/SASSYS-1 code system has been copyrighted by ANL and its license is handled by Technology Development & Commercialization Division. ANL will assist in the licensing agreement, including export control procedures.

Deliverables:
Report on identification and status of critical experiments data (1.31)
Summary of modeling features and experimental data used for the formulation of SAS4A/SASSYS-1 fast reactor safety analysis code (12.31)

[redacted]

3. IGCAR(인도) 기술협력 현황 및 계획

○ 목적

- 현재 SFR 분야에서 가장 활발한 연구개발 활동을 진행하고 있는 인도와의 기술협력을 통해 국내 개발 SFR의 기술수준 향상, 비용 절감 및 일정을 단축하고자 함

○ 경과

- KAERI와 IGCAR 재료분야 MoU 기체결(‘12.1)
- 상호 기술분야 협의(‘12.11)
- 사업단과 IGCAR SFR 안전성 분야에서 MoU 체결(‘13.1)

○ 향후 계획

- 인도와의 협력은 국제적으로 매우 민감한 사항이므로 조심스럽게 접근할 것임
- 우선 소듐기술 등 기초기술과 안전성 증진 분야의 협력을 중심으로 추진할 것임

○ 협력 현황

- 현재 SFR 개발 분야에 가장 활발한 연구개발 활동을 진행하고 있는 인도 IGCAR과의 기술협력을 통하여 국내 개발 SFR의 기술수준을 향상시키고, 비용을 절감함은 물론 개발 시기를 앞당길 수 있을 것임
  · IGCAR은 실험로 FBTR을 운전중임
  · 원형로인 PFBR을 건설을 완료하고 금년 중으로 운전 준비에 들어갈 예정임
  · 금속연료 장전 노심인 MFTR을 설계하고 있으며 2017년 건설시작 예정임
  · ‘23년경 상용로 CFBR(500MWe) 건설완료 예정임
  · ‘25년 이후 FFBR(1,000MWe) 건설 계획을 가지고 있음
  · 현재 SFR 개발 분야에서 가장 활발한 연구개발 활동을 수행하고 있음
- KAERI와 IGCAR이 체결한 재료분야의 MoU를 바탕으로 SFR 분야에도 고온건전성 연구의 상호협력을 추진중에 있었음
  · KAERI와 IGCAR이 재료분야의 MoU를 바탕으로 SFR 분야에도 상호 세미나 개최 등을 통하여 지속적으로 상호 협력을 논의하여 옴
  · 2012년 11월 17일~21일에는 사업단 2인과 차세대핵연료개발부 1인이 IGCAR을 방문하여 기술토의를 진행하고 및 실험시설을 견학한 바 있음
- 사업단에서는 SFR 분야의 본격적인 협력을 위하여 별도의 MoU를 요청하였으며, 상호 충분한 협의를 통하여 2013년 1월 25일 MoU (SFRA-IGCAR MoU in the area of

SFR Safety)를 체결하였음[표 6]

· MoU의 주요 내용은 기존 재료분야의 MoU를 기반으로 하여 SFR Safety 분야로 대체한 것임

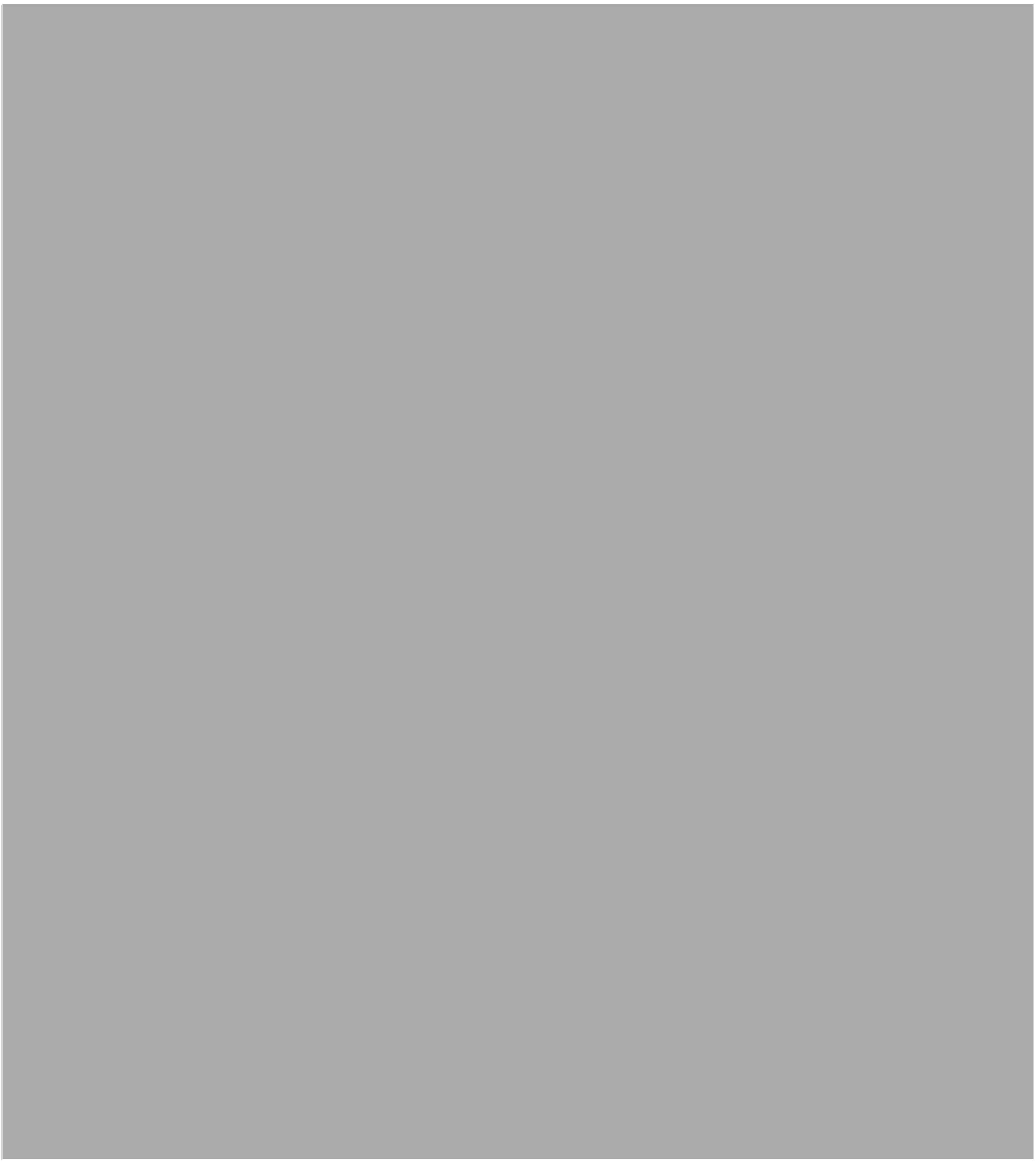
– 향후계획

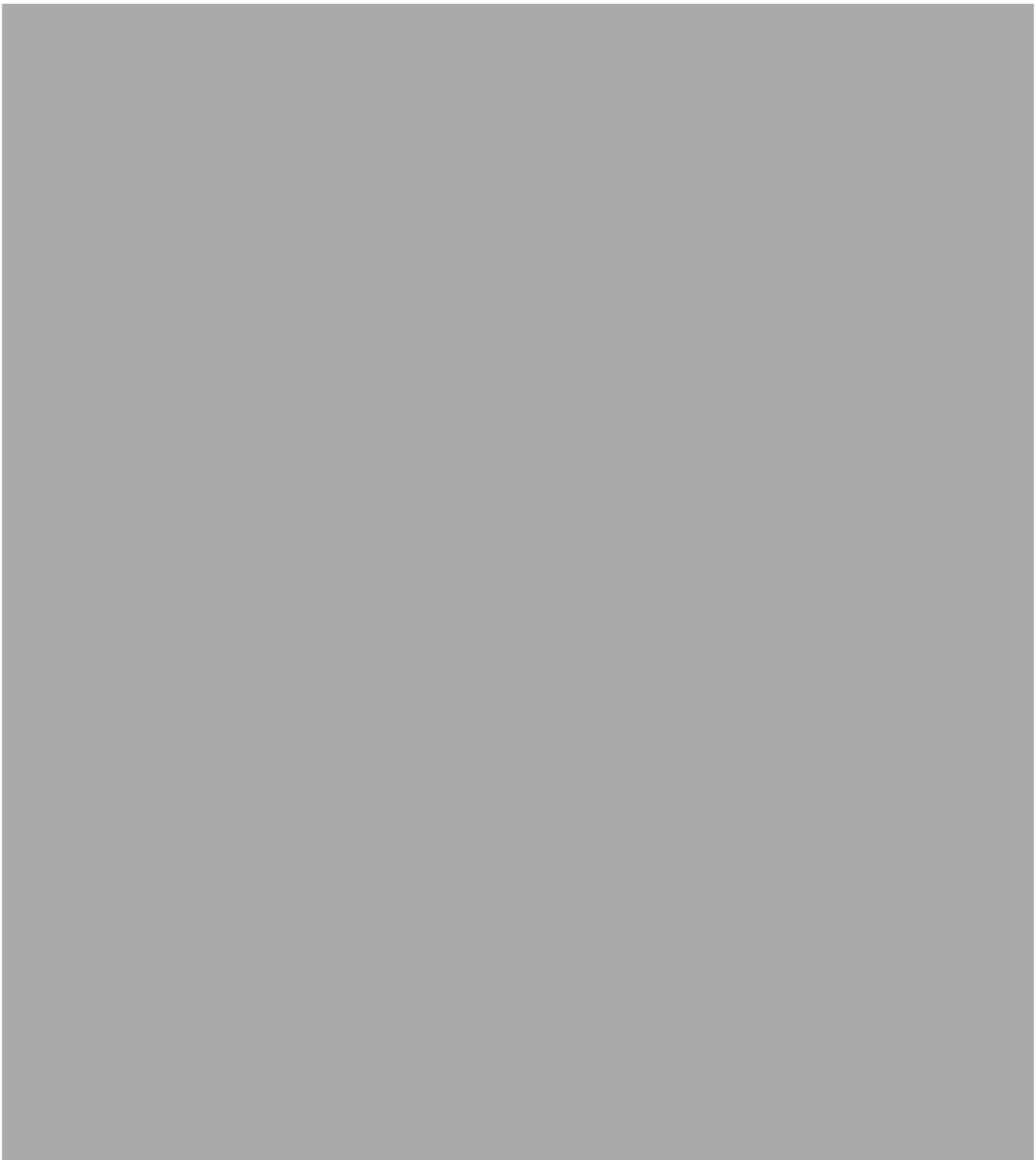
· 인도와의 협력은 국제적으로 매우 민감한 사항이므로 조심스럽게 접근할 것이며, 우선 소듐기술 등 기초기술과 안전성 증진 분야의 협력을 중심으로 추진할 예정임

· 인도의 연구자를 전문가로 초빙하여 우리의 기술개발에 대한 기술 자문 받을 계획을 추진할 예정임

그림 7. SFRA-IGCAR MOU 서명식

표 6 SFRA-IGCAR MOU





4. 제1회 국제 전문가 기술 검토회의

○ 목적

- 예비 개념설계 결과에 대한 검토를 통하여 문제점 파악 및 개선방안 도출
- 국제 전문가 토의를 통해 SFR 원형로 주요 기술현안 해결 방안 모색

○ 회의 일정표[표 7]

- 2012년 1월 23일(수)~25일(금) 3일간 표 7과 같은 일정으로 회의 진행

○ 국외 초청 전문가 (총 8인: 미국 1인, 일본 2인, 중국 2인, 인도 2인, 프랑스 1인)

○ 회의 검토 의견[표 8]

- 사업 계획이 약간은 느슨한 듯 보이므로, 전체적인 개발 일정 및 실험계획 등을 공정계획을 통하여 더욱 구체화하는 것이 필요함
- 최적화된 PGSFR 개발을 위해서는 각 분야 간 적절한 설계 연계가 더욱 긴밀하게 이루어져야 할 것임
- 상세설계 착수 전에 사업 초기 단계부터 규제 기관과의 긴밀한 논의를 통하여 필요한 사항을 사전에 확정하는 것이 필요함
- 3일간의 회의를 통하여 유익한 검토의견을 청취할 수 있었으며, 그 내용은 참가자들의 검토를 거쳐 표 8과 같이 정리하고 서명을 받음
- 향후 사업 방향 설정 또는 과제 기획시 반영하도록 할 예정

○ 향후계획

- 금번과 같은 국제 전문가 기술검토 회의를 년 2회 개최
- 우리의 기술 개발 방향을 점검하고 부족한 부분에 대한 해외협력 방안을 도출함으로써,

국내 SFR 개발 기간과 투자비용 절감

그림 8. 제1회 국제 전문가 기술 검토회의 참가자

표 7 제1회 국제 전문가 기술검토 회의 일정표

The 1st International Technical Review Meeting For Prototype Gen IV SFR(PGSFR) Development in Korea

23rd ~ 25th January 2013
INTEC Conference Room(2F), KAERI, Daejeon, Korea

FINAL AGENDA

23rd January 2013 (Wednesday)

~	Arrival and Reception
09:15	**Opening Address (Purpose of the meeting)** - Mr. Won-Seok PARK, Director of SFRA/KAERI
09:20	**Introduction of Invited Participants & Review of Agenda** **Nomination of Honorary Chair, General Chair and Session Chairs** **Honorary Chair: Mr. Subash Chander CHETAL** **General Chair: Mr. Won Sik YANG**
09:25	**PGSFR Program and Technical Issues** - Mr. Hyung-Kook JOO, Technical Manager of SFRA/KAERI
09:45	**PGSFR Design Development, Review and Discussion** - Mr. Yeong-Il KIM, Manager of SFR Design Division, KAERI
12:00	Lunch
13:30	**V&V Studies for PGSFR, Review and Discussion** - Mr. Yong-Bum LEE, Manager of SFR V&V Division, KAERI
15:30	Coffee Break
15:45	**Metal Fuel Development for PGSFR, Review and Discussion** - Mr. Chan-Bock LEE, Manager of Metal Fuel Development Division, KAERI
17:45	Adjourn
18:30	Dinner

Presentations of SFR project status of each country will be flexibly arranged within the schedule

- 1 -

24th January 2013 (Thursday)

09:15 **Discussion 1 (Steam Generator)**
Chairs: Mr. Subash Chander CHETAL, Mr. Masakazu ICHIMIYA
Technical issues presentation: Mr. Gyeong-Hoi KOO
Discussion Topics
- Past steam generator operating experiences and improvement in design
- Counter measure against sodium-water reaction in SG
- Single or double wall tube SG design
- Monolithic or modular type of SG design and its influences on reliability and plant economy
- Recent manufacturing technology of SG and its implication on reliability and plant economy

12:00 Lunch

13:30 **Discussion 2 (Severe Accident)**
Chairs: Mr. Donghui Zhang, Mr. Perumal CHELLAPANDI
Technical issues presentation: Mr. Hae-Yong JEONG
Discussion Topics
- Recent licensing approaches of severe accident after Fukushima accident
- Characteristics of severe accident propagation in metal fueled core
- Severe accident mitigation system design

15:30 Coffee Break

15:45 **Discussion 3 (S-CO2 Brayton Cycle)**
Chairs: Mr. Alfredo VASILE, Mr. Yong-Hee KIM
Technical issues presentation: Mr. Yong-Hee KIM
Discussion Topics
- Innovative heat transport system design against Na-water reaction
- Supercritical CO_2 Brayton cycle as an alternative power generation cycle
- Technical issues to be solved for application of S-CO2 Brayton cycle to SFR

17:45 Adjourn

18:30 Dinner

25[h] January 2013 (Friday)

09:15 **Discussion 4 (RHRS)**
Chairs: Mr. Kazumi AOTO, Mr. Yizhe LIU
Technical issues presentation: Mr. Tae-Ho LEE
Discussion Topics
- Residual decay heat removal system design
- Trade-off among different concepts of RHRS (hot pool cooling, vessel cooling, steam generator cooling concepts)
- Optimal design concepts of RHRS for PGSFR

12:00 Lunch

13:30 **Discussion 5 (General Discussion for the Implementation of PGSFR)**
Chairs: Mr. Won-Seok PARK, Mr. Subash Chander CHETAL
Technical issues presentation: Mr. Won-Seok PARK
Discussion Topics
- Overview & discussions
- Licensing legislation and procedure for the prototype reactor
- Enhancement of bilateral cooperation
- Signature of meeting minute and next meeting

15:30 Wrap-up & Coffee Break

Group 1 (Technical Discussions)

15:40 Free Discussions for SFR Development
- Free Discussions

17:00 Adjourn

18:00 Dinner

Group 2 (Technical Tour)

15:40 Security Check at the entrance to KAERI

15:45 **Technical Tour (STELLA-1)**

16:30 **Technical Tour (PRIDE)**

17:00 Adjourn

18:00 Dinner

■ **Oversea Invited Experts:**

[redacted]

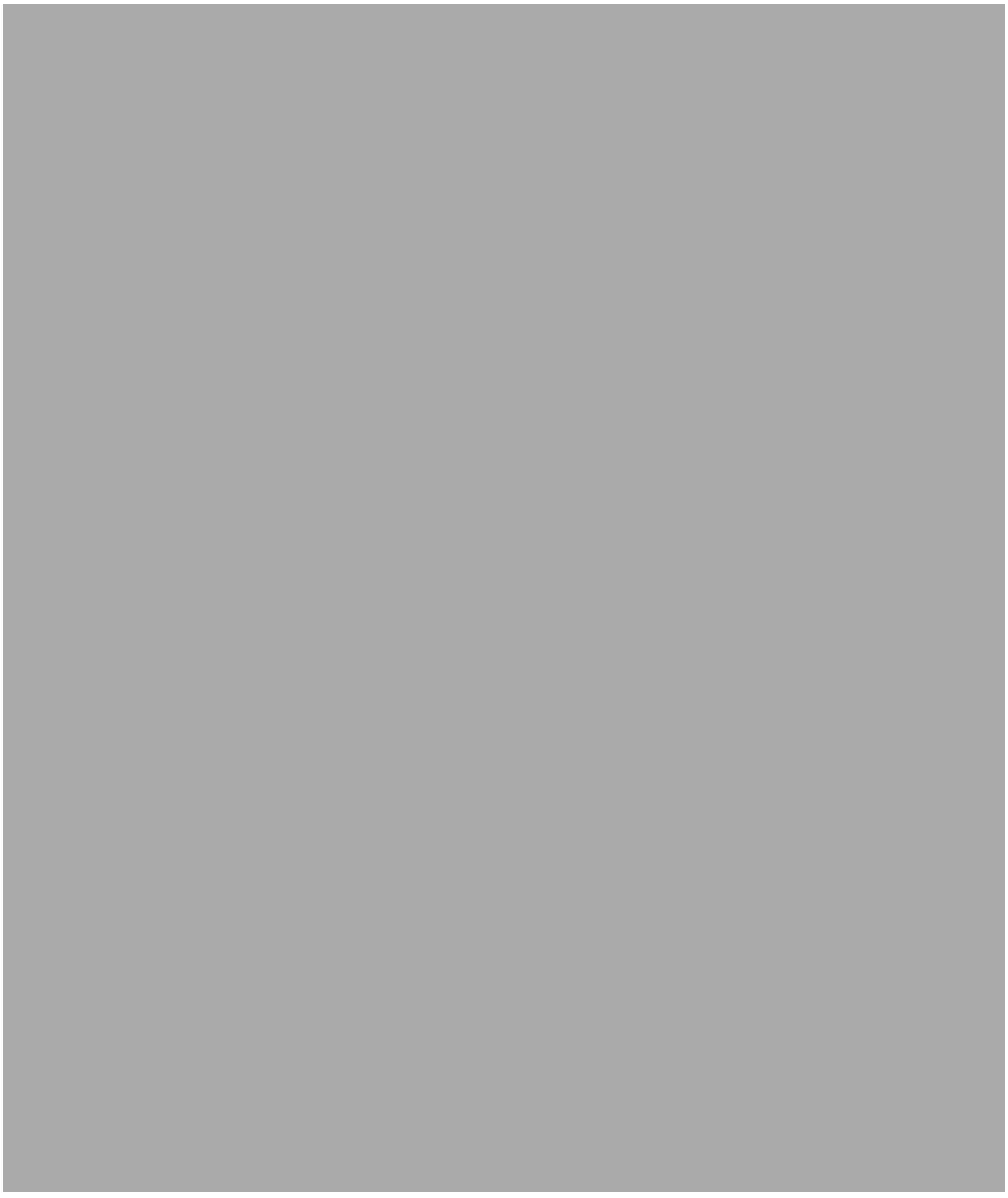
■ **Special Invitee:**

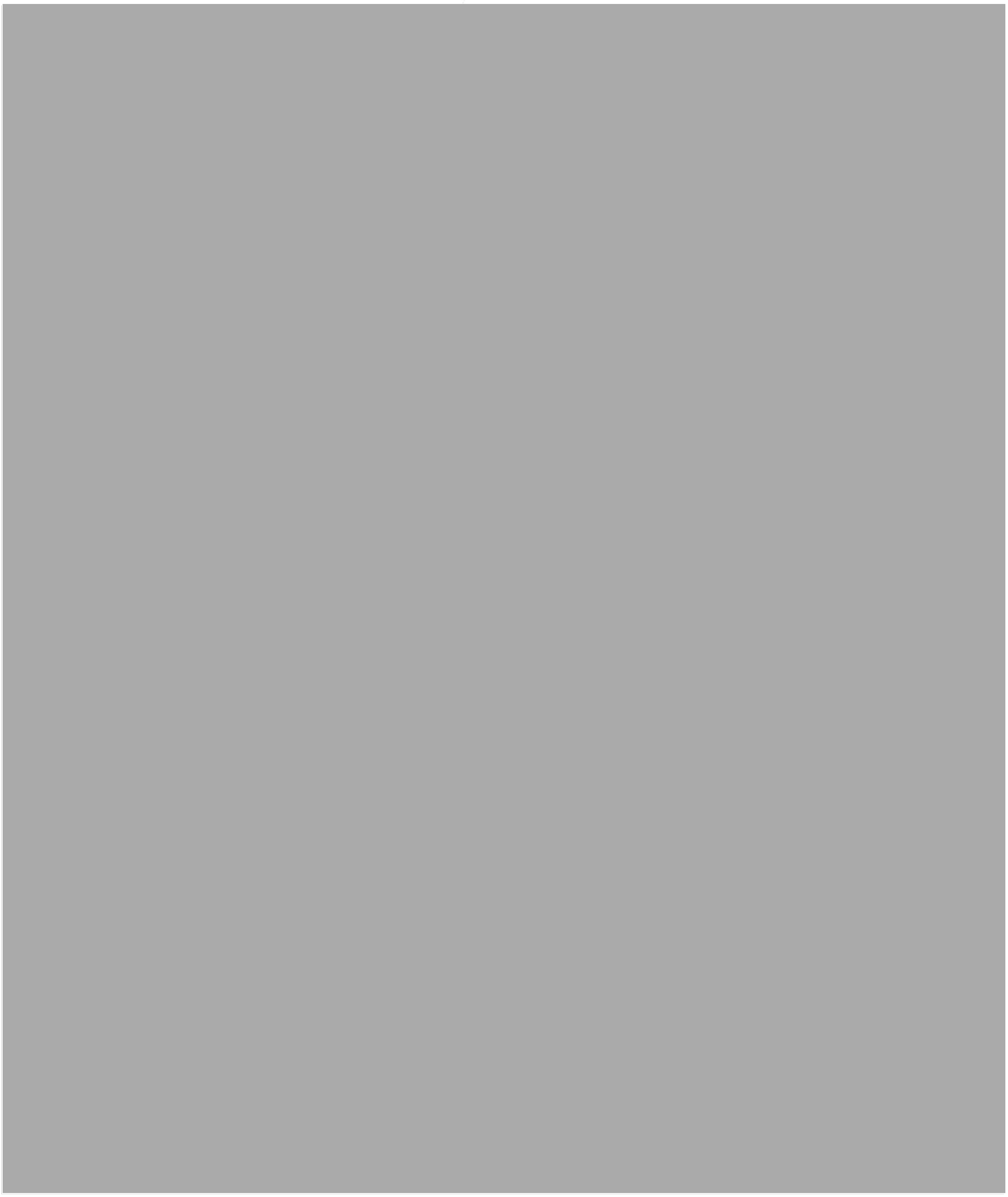
[redacted]

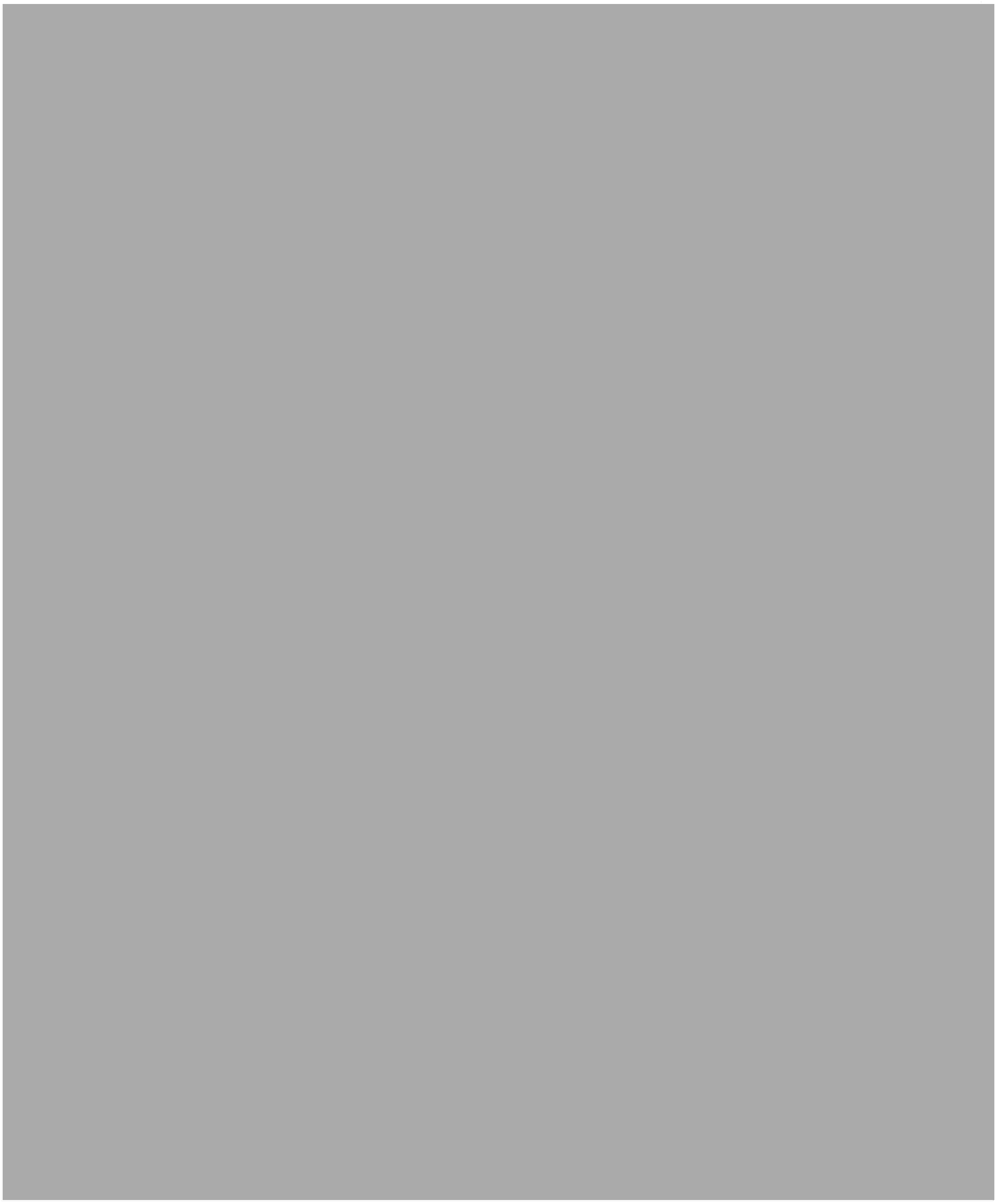
■ **KAERI Participants:**

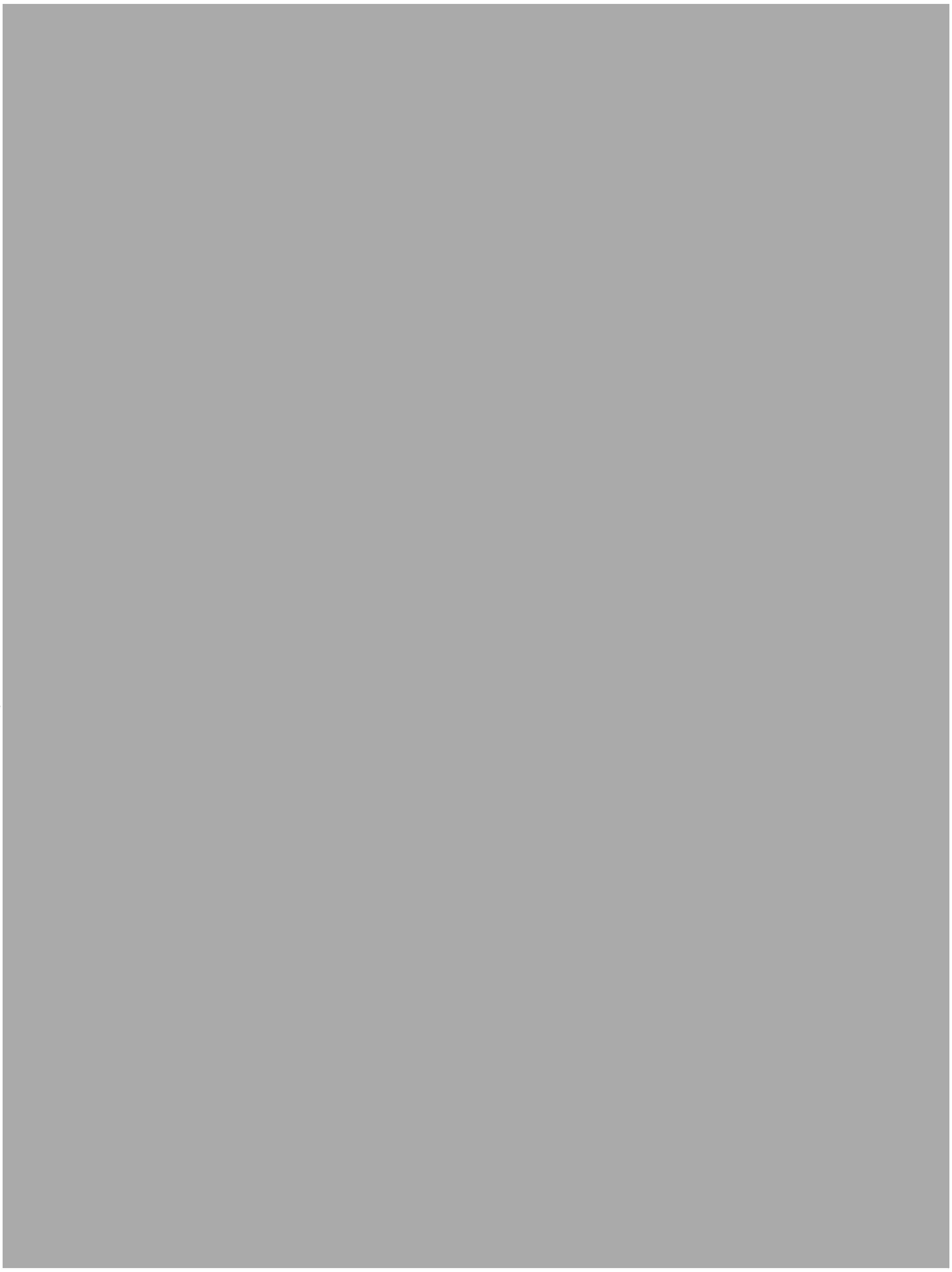
[redacted]

표 8　　제1회 국제 전문가 기술검토 의견









5. 원형로 특정설계 인허가 방안 구축

○ 목적

- 현재의 인허가 절차에 따를 경우 기술검증을 위해 건설하는 원형로의 인허가는 매우 어려울 것으로 판단됨
- 따라서, 규제기관으로 하여금 원형로의 특정설계에 대한 인허가를 규정하는 법안의 필요성을 수용하도록 하는 업무가 필요함

○ 경과

- 원자력안전기술원의 규제개발 과제 참여자들과의 지속적인 업무 협의
- 원자력안전위원회 담당부서와 SFR의 기술특성과 개발 일정 등에 대한 세미나 진행
- 규제법의 필요성에 대하여 원자력안전위원회 담당부서와의 지속적인 업무협의

○ 향후 계획

- 정부조직법 개정과 함께 조직 개편과 담당업무 변경 등으로 인하여 업무협의 진행이 미비한 상태에 있음
- 가칭 원자력 안전 검토 위원회 구성 등을 통한 지속적인 업무협의 예정
- 원자력안전위원회, 원자력안전기술원 외에 원자력 분야 전문가들의 공감대 형성을 위한 노력 병행 예정

6. 소듐냉각고속로 원형로 개발을 위한 예비타당성 분석 및 에너지 MIX 옵션 스터디

6-1. 제4세대 소듐냉각고속로 기술개발사업 예비타당성조사 사전 대응 연구

○ 연구내용

- 소듐냉각고속로 개발 타당성 제고 방안 도출 및 사업방향 설정

○ 위탁과제 책임자

- 김지용 (기술과가치)

○ 연구목적

- 소듐냉각고속로 개발 타당성 제고 방안 도출 및 사업방향 설정을 위하여 제4세대 소듐냉각고속로 기술개발사업 예비타당성조사 사전 대응 연구를 위탁연구로 진행함

○ 연구결과

- 제4세대 소듐냉각고속로 기술개발사업 예비타당성조사 사전 대응 연구는 위탁연구보고서로 제출되었음[보고서 번호: KAERI/CM-1772/2012]

○ 향후 계획

- 제4세대 소듐냉각고속로 기술개발사업 예비타당성조사 사전 대응 연구를 통하여 도출된 내용을 사업단 운영에 반영하고 보완할 예정임

6-2. 초임계 CO_2 브레이튼 사이클 SFR 적용 타당성 연구

○ 연구내용

- 소듐냉각고속로에 적용가능한 초임계 이산화탄소 싸이클을 포함한 다양한 폐회로 브레이튼 싸이클에 대한 검토

○ 위탁과제 책임자

- 이정익 (KAIST 원자력 및 양자공학과)

○ 연구목적

- 기존의 증기 랭킨 싸이클에 비해서 운전범위도 넓고, 열효율도 높으며, 싸이클 전체 부피가 작고, 재료 문제도 비교적 완화되기 때문에 다양한 장점을 보유하고 있는 초임계 CO2 브레이튼 사이클 SFR 적용 타당성 연구를 위탁연구로 진행함

○ 연구결과

- 기존의 증기 랭킨 싸이클에 비해서 운전범위도 넓고, 열효율도 높으며, 싸이클 전체 부

피가 작고, 재료 문제도 비교적 완화되기 때문에 다양한 장점을 보유하고 있는 초임계 CO2 브레이튼 사이클 SFR 적용 타당성 연구 결과는 위탁연구보고서로 제출되었음[보고서 번호: KAERI/CM-1773/2012]

○ 향후 계획

- 초임계 CO2 브레이튼 사이클 SFR 적용 타당성 연구는 범위를 확대하여 진행할 예정임

6-3. 녹색에너지시대의 장기 에너지 Mix 검토

○ 연구내용

- 국가 미래 장기에너지 수요전망, 에너지원별 수급문제 분석을 통하여 국내 장기 에너지 수급방안과 원자력발전에 대한 정책 방안

○ 위탁과제 책임자

- 신정식 (중앙대학교)

○ 연구목적

- 국가 미래 장기에너지 수요전망과 에너지원별 수급문제 분석하여 국내 장기 에너지 수급방안과 원자력발전에 대한 정책 방안을 위탁연구로 진행하여 제시함

○ 연구결과

- 국가 미래 장기에너지 수요전망과 에너지원별 수급문제 분석하여 국내 장기 에너지 수급방안과 원자력발전에 대한 정책 방안은 위탁연구보고서로 제출되었음[보고서 번호: KAERI/CM-1774/2012]

○ 향후 계획

- 국내 장기 에너지 수급방안과 원자력발전에 대한 정책 방안으로 도출된 내용을 토대로 SFR의 필요성을 적극 홍보할 예정

제 4 장 목표 달성도 및 관련분야에의 기여도

제1절 세부목표대비 주요 성과 및 달성도

세부목표	주요 성과	달성도
해외협력 및 원형로 특정설계 인허가 준비	- 소듐냉각고속로 원형로 설계를 위한 해외협력 방안 구축 - 미국 ANL과의 기술협력을 통한 개념설계 Review - 소듐냉각고속로 원형로 특정설계 인허가 기반 마련 방안 구축	100%
예비타당성분석 및 옵션 스터디	- 제4세대 소듐냉각고속로 기술개발사업을 위한 예비타당성 분석 및 옵션 스터디	100%

제2절 관련분야에의 기여도

본 과제는 직접적인 연구과제의 성격보다는 연구과제의 효율적인 목표 달성을 위한 해외협력과 원형로 인허가 등에 대한 사전 대비를 위한 성격을 갖고 있다. 따라서 적절하고 효율적인 해외협력과 국제전문가 자문회의 및 인허가 대비 검토 등을 통해 연구과제를 효율적으로 운영하도록 하는데 초점이 맞춰져 있다. 본 연구를 통해 2013년 소듐냉각고속로개발사업의 성공을 위한 해외협력 방안 마련과 인허가 대비를 위한 연구를 수행하였다.

(1) 기술적 측면

- 해외 협력을 통하여 국내의 독자적인 설계 및 해석 기술 능력을 향상시켜 적기에 특정설계승인 획득할 수 있도록 함
- 소듐냉각고속로 설계 등 기술 개발 경험이 풍부하며 특히 금속연료 기술 및 관련 데이터를 확보하고 있는 미국 ANL과의 협력을 통하여 국내 금속연료 기술 개발에 활용함
- 원형로 특정설계 인허가 업무를 위한 기반을 조성하고 규제기관으로 하여금 소듐냉각고속로 원형로 특정설계 인허가에 대한 대비를 세우도록 함
- 초임계 CO2 브레이튼 사이클 SFR 적용 타당성 연구를 통하여 기존의 소듐냉각고속로는 근본적으로 해결이 어려운 소듐-물 반응이 발생하지 않는 계통 구성 가능성을 검토하여 안전성이 향상된 SFR 계통 구성 가능성을 검토함

(2) 경제적·산업적 측면

- 해외 협력을 통하여 국내의 설계 및 해석 기술 능력을 향상시켜 시간과 비용을 절감할 수 있도록 함
- 미국 ANL과의 협력을 통하여 국내 금속연료 기술 개발 비용과 시간을 절약할 수 있도록 함
- 원형로 특정설계 인허가 업무 기반 조성으로 소듐냉각고속로 원형로 특정설계 인허가 업무 효율성 제고하도록 함
- 원자로 계통의 소형화 및 모듈화만 생각하는 것이 아니라 전력변환 계통의 소형화 및 모듈화에 대한 고려도 함께 하는 것이기 때문에 소듐냉각고속로의 경제성 확보에 기여할 것으로 예상됨

제 5 장 연구개발결과의 활용계획

해외협력 방안 구축을 통하여 채득한 방안에 따라서 향후 협력을 진행함으로서 국내 기술개발의 효율성을 제고하고 국내 소듐냉각고속로 건설 목표를 달성할 수 있도록 활용할 것이다. 또한, 개념설계 검토 및 기술 개발 경험을 이용하여 국내 원형로 설계의 신뢰도를 향상시킬 수 있도록 활용할 예정이다.

인허가 방안 구축을 통하여 국내 원형로 특정설계 인허가 법규를 정비하고 규제기관으로 하여금 인허가 업무를 효율적으로 진행할 수 있는 기반을 조성하고, 소듐냉각고속로 개발 사업의 사전 예비타당성을 평가함으로서 향후 정부의 예비타당성 평가에 대비할 것이다.

초임계 CO_2 브레이튼 사이클 SFR 적용 타당성 연구를 통하여 근본적으로 소듐-물 반응으로부터 자유로운 계통 구성 가능성을 검토하여 안전성과 경제성이 향상된 SFR 계통 구성 타당성을 검토할 것이다.

제 6 장 연구개발과정에서 수집한 해외 과학기술 정보

해당사항 없음

제 7 장 연구시설 · 장비 현황

해당사항 없음

주 의

1. 이 보고서는 교육과학기술부에서 시행한 원자력연구개발사업의 연구보고서입니다.
2. 이 보고서 내용을 발표하는 때에는 반드시 교육과학기술부에서 시행한 원자력연구개발사업의 연구결과임을 밝혀야 합니다.
3. 국가과학기술 기밀유지에 필요한 내용은 대외적으로 발표하거나 공개하여서는 아니됩니다.

연구성과(연구사업지원시스템 입력성과)

※ 연구사업지원시스템(http://ernd.nrf.re.kr)을 참고하여 입력된 성과 다운받아 작성

사업명	소듐냉각고속로개발사업단	연구책임자	김영균	주관기관	한국원자력연구원
과제번호	2012M2A8A2054770	과제명	소듐냉각고속로 해외협력 및 인허가 방안 구축		

과학기술/학술적 연구성과(단위 : 건)

전문학술지 논문게재				초청강연실적	학술대회 논문발표		지식재산권				수상실적	출판실적	
국내논문		국외논문			국내	국제	출원		등록			저역서	보고서
SCI	비SCI	SCI	비SCI				국내	국외	국내	국외			
0	0	0	0	0	0	0	0	0	0	0	0	0	0

인력양성 및 연구시설(단위 : 명,건)

학위배출		국내외 연수지원				산학강좌	연구기자재
박사	석사	장기		단기			
		국내	국외	국내	국외		
0	0	0	0	0	0	0	0

국제협력(단위 :명,건)

과학자교류		국제협력기반			학술회의개최	
국내과학자 해외파견	외국과학자 국내유치	MOU체결	국제공동연구	국제사업참여	국내	국제
0	0	2	0	0	0	1

산업지원 및 연구성과 활용(단위 : 건)

기술확산				연구성과활용(사업화 및 후속연구과제 등)			
기술실시계약	기술이전	기술지도	기술평가	후속연구추진	사업화추진중	사업화완료	기타목적활용
0	0	0	0	0	0	0	0